EA-90-01

ENVIRONMENTAL ASSESSMENT BOARD

Ontario

ONTARIO HYDRO DEMAND/SUPPLY PLAN HEARINGS

VOLUME: 121

DATE: Tuesday, March 24, 1992

BEFORE:

HON. MR. JUSTICE E. SAUNDERS	Chairman
DR. G. CONNELL	Member
MS. G. PATTERSON	Member

FARR & ASSOCIATES REPORTING INC.

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EA-90-01

ENVIRONMENTAL ASSESSMENT BOARD
ONTARIO HYDRO DEMAND/SUPPLY PLAN HEARING

IN THE MATTER OF the Environmental Assessment Act, R.S.O. 1980, c. 140, as amended, and Regulations thereunder;

AND IN THE MATTER OF an undertaking by Ontario Hydro consisting of a program in respect of activities associated with meeting future electricity requirements in Ontario.

Held on the 5th Floor, 2200 Yonge Street, Toronto, Ontario, on Tuesday, the 24th day of March, 1992, commencing at 10:00 a.m.

VOLUME 121

B E F O R E :

THE HON. MR. JUSTICE E. SAUNDERS	Chairman
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MS. G. PATTERSON	Member

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---Upon commencing at 10:00 a.m.

THE REGISTRAR: Please come to order. This hearing is now in session. Please be seated

THE CHAIRMAN: For the purpose of the record, I would like to record some exhibits that have been filed since we last met.

Exhibit 449A, NAN Treaty #3, Teme-Augama, addendum to the memorandum of understanding, Exhibit 449.

452A by Ontario Hydro, Data for Figures of Exhibit 452, Demand/Supply Update, 1992.

That document was presented earlier in the hearing but there was no formal recording of its introduction as an exhibit.

Exhibit 518, Professor R.E. Munn, Climate Change and annotated Bibliography.

The parties will recall that Professor Munn is a consultant retained by this panel, and this document is filed in accordance with respect to expert witnesses that we dealt with earlier in the hearing.

Exhibit 519, Ontario Hydro filing, Nuclear Panel Overheads which will be used in the Hydro evidence in chief.

Exhibit 520, which are the Panel 9 interrogatories number. There are 29 of those

interrogatories already put in by Ontario Hydro and they are listed and that list is available.

Exhibit 521 filed by IPPSO, entitled Evaluating the Premature Retirement of Nuclear Facilities, A Case Study.

Exhibit 522, also filed by IPPSO, Canadian Nuclear Association brief to the Standing Committee on Energy, Mines and Resources.

Those are the exhibits which I have been advised that have been filed since we last meet, on the 10th of March.

---EXHIBIT NO. 449A: Addendum to the memorandum of understanding, Exhibit 449, filed by NAN Treaty #3, Teme-Augama.

---EXHIBIT NO. 452A: Data for Figures of Exhibit 452, Demand/Supply Update, 1992, filed by Ontario Hydro.

---EXHIBIT NO. 518: Climate Change and annotated Bibliography, by Professor R.E. Munn.

---EXHIBIT NO. 519: Nuclear Panel Overheads.

---EXHIBIT NO. 521: Document entitled, Evaluating the Premature Retirement of Nuclear Facilities, A Case Study, filed by IPPSO.

---EXHIBIT NO. 522: Canadian Nuclear Association brief to the Standing Committee on Energy, Mines and Resources, filed by IPPSO.

THE CHAIRMAN: Now, Ms. Harvie, are you ready to proceed?

Are there any scoping problems that need

to be dealt with.

MS. HARVIE: Well, I don't have any submissions, unless anyone else does, Mr. Chairman.

THE CHAIRMAN: Thank you.

Anyone else?

Yes, sir? You are...

MR. WRIGHT: Timothy Wright.

THE CHAIRMAN: Yes, Mr. Wright.

MR. WRIGHT: I am not sure that this is the right kind of problem or question, but I have been talking to Ms. Harvie about the witnesses that are going to be available for this panel, and my understanding is that there are none from the human resource side of Ontario Hydro.

What we are dealing with is safety and safety is a combination of the technical expertise of the systems and the motivation of the people. I think we need some human resource people so that they can be questioned as to the level of motivation and moral and the systems for monitoring that.

THE CHAIRMAN: Anyone else want to make any submission in support of that?

I think the short answer to that, Mr. Wright, is that it is up to Hydro to decide who they call as witnesses. If they don't call the right

witnesses, that is a matter that will, if that's true, weaken their case. That's up to them. They have the choice of what witnesses to call.

MR. WRIGHT: Thank you.

THE CHAIRMAN: Mr. Greenspoon?

MR. GREENSPOON: Yes, Mr. Chairman, this is probably a matter that should better be raised before cross-examination begins, but I wish to put the Board and Hydro and other parties on notice about this issue, and that is Northwatch is very concerned about, for lack of a better phrase, the pro-nuclear components, splitting their cross-examination. It appears as though there is a financial relationship between --

THE CHAIRMAN: I'm sorry, what do you mean by the pro-nuclear, splitting their cross-examination?

MR. GREENSPOON: Well, Atomic Energy of Canada Limited, the Canadian Nuclear Association, CANDU Industries and Ontario Hydro all have an interconnected financial relationship. It's very clear that Ontario Hydro spends money.

THE CHAIRMAN: Just a minute. I want to understand what your problem is. Ontario Hydro's evidence is going in first, followed by AECL, followed

by the Canadian Nuclear Association.

MR. GREENSPOON: The issue I wish to raise is, should the Canadian Nuclear Association have the right to cross-examination, given their connection with the Atomic Energy of Canada Limited.

All I am asking is - it's a rhetorical question to some extent - that what I would like on behalf of my clients, before the cross-examination begins, I think it is useful for all of the parties at this hearing to know what the relationship between all of these parties is, so that we can fairly look at and understand where they are all coming from in their cross-examination. Otherwise, what you have is a party who really the same party cross-examining twice and, whether they follow each other or not, I don't think that that is really proper.

THE CHAIRMAN: We have parties with similar and sometimes identical interests who have cross-examined in the first eight or nine panels. That's never been a problem that's been raised before.

MR. GREENSPOON: No, I think the difference here is that there is a financial connection between the parties.

THE CHAIRMAN: I think that's a matter of evidence rather than the right to cross-examine.

MR. GREENSPOON: I wish to put that on the record. Thank you.

THE CHAIRMAN: Mr. Poch?

MR. D. POCH: Mr. Chairman, just a similar concern to Mr. Greenspoon's, it's slightly different though. The Canadian Nuclear Association is in a very different position, that is an organization who has as its members, CANDU Industries, AECL and Ontario Hydro, and it's a matter of public information that most of their funding comes from AECL and Ontario Hydro.

So we have an organization which Hydro a member purporting to cross-examine Ontario Hydro. Indeed, we have them following, and if the indication this morning is correct, planning to spend I think it was a day-and-a-half cross-examining after AECL plans to spend two to four days cross-examining. I don't propose that they be denied the right to ask questions because they may have interests of other members beyond the interests that Ontario Hydro shares. But I think it would be very appropriate for them to at least voluntarily and if necessary for the Board to exercise control over the kind of redundancy we might expect and the abuse of cross.

THE CHAIRMAN: We will have to see when

that arises. If that happens then we will have to deal with it. We can't anticipate that it is going to arise.

MR. D. POCH: No, I just felt it was important that the Board be aware of the fact that the CNA has as a principal member Ontario Hydro, that puts them in an unusual position.

THE CHAIRMAN: Anything else?

All right, Ms. Harvie, are you ready to start your examination?

MS. HARVIE: I am, Mr. Chairman.

THE CHAIRMAN: There is a motion, isn't there, by Energy Probe; is that correct, Mr. Mattson?

MR. MATTSON: Mr. Chairman, that's scheduled for tomorrow at the end of evidence in chief by Ontario Hydro.

THE CHAIRMAN: It's to follow Hydro's evidence in chief. That's the understanding?

MR. MATTSON: Yes.

THE CHAIRMAN: Thank you.

MS. HARVIE: Yes, Mr. Chairman, there was a letter of March 18th addressed to Mr. Nunn. There are additional copies here. This shows the interrogatories in the same order in which they will be referred to by the witnesses in their evidence in

chief, so rather than have them pause and say this has been assigned Interrogatory No. 520 blank blank, it's all in the list in the same order. So I can perhaps give you this list so that you can follow it as we are going through the evidence, and I will give additional copies to Mr. Lucas as well.

THE CHAIRMAN: As a matter of fact, I think we have a list. It runs through to No. 29; is that correct?

MS. HARVIE: Yes, that's correct.

We also have rapidly depleting copies of the correspondence that has been sent out, witness CVs, the list of interrogatories as I mentioned, materials relating to health effects and the overhead package is all gone. If people need additional copies we will certainly provide them.

In addition to that --

THE CHAIRMAN: You are referring to Exhibit 507 and 519?

MS. HARVIE: Yes, that's correct, Mr. Chairman.

In addition to that I would like to file an errata to Exhibit 507. This was brought to our attention yesterday by Mrs. Mackesy. I am very grateful for it. I will leave it right here, perhaps

that's the simplest way of handling it.

THE CHAIRMAN: What is that?

MS. HARVIE: This is an errata to Exhibit 507 referred to as the health report. Perhaps this should be given its own exhibit number.

THE CHAIRMAN: It should have its owns exhibit number.

THE REGISTRAR: No. 523.

---EXHIBIT NO. 523: Health Report.

[10:12 a.m.]

THE CHAIRMAN: Did you save three for us?

MS. HARVIE: No. If people would like additional sets of overheads perhaps they could wave their hands and we could send a message back to the photocopiers in the back room to get to work. One, two, three, four? All right. Thank you.

Mr. Chairman, before having the witnesses sworn I would like to introduce the witnesses to the Board and the parties and outline, very briefly, the areas that each will be responsible for so that you have some sense of what the oral evidence will cover.

The witness closest to the counsel table is Dr. David Whillans, W-h-i-l-l-a-n-s, and Dr. Whillans is a Senior Safety Specialist with the Science and Technology Department of the Health and Safety

Division, and he will be addressing the health effects of ionizing radiation exposure. He will be describing the exposure limits and Hydro's occupational and public health performance.

Seated next to Dr. Whillans is Mr. Kurt Johansen. Mr. Johansen is a Supervisor with the Environmental Studies and Assessment Department, Design and Construction Branch, and he is here to speak to the natural environmental effects of nuclear generation, and he will be describing specifically environmental regulation and performance and how Hydro manages radioactive materials, particularly used fuel and low and intermediate waste.

Seated to the left of Mr. Johansen is Mr. Frank King. Mr. King is the Section Head of the Risk Assessment Section of the Nuclear Safety Department, also of the Design and Construction Branch. He will be giving evidence on nuclear safety generally and specifically on how safety is managed and regulated. In addition to that, he will be giving evidence on Hydro's safety performance, and lastly, he will be describing the emergency planning in the event of an accident.

To the left of Mr. King is Mr. William Penn. Mr. Penn is the Program Manager, Generation

Planning and Approvals Group of the Design and Construction Branch. Mr. Penn will provide an overview of the current nuclear program, and he will also be speaking on the current and future costs associated with the existing system, and lastly, he will be describing some different nuclear options that were considered for analysis purposes for Exhibit 452, the Updated Demand/Supply Plan.

Finally, the witness seated closest to the Panel is Mr. Ian Daly. Mr. Daly is a Section Head in the Nuclear Operation Standards Department, the Nuclear Operations Branch, and he will be addressing the operational performance of the existing system in the past, present and future.

May I ask that the witnesses be sworn in, Mr. Lucas?

THE REGISTRAR: Will you please stand?

DAVID WHILLANS,
KURT JOHANSEN,
FRANK CALVIN KING,
WILLIAM JOHN PENN,
IAN NICHOL DALY; Sworn.

THE REGISTRAR: Thank you, gentlemen.

MS. HARVIE: Mr. Chairman, the order of evidence is the same as that set out in the Statement of Proposed Issues. This is again a letter to Ms. Morrison, dated March the 13th. If you don't have

copies of that I can give you copies, but it may help you in following which segments we are going through.

Just before we start, the mounted board there to your left is an excerpt from the overhead package. I think it's page 62 of Exhibit 519. The replica of that is in the package of materials.

DIRECT EXAMINATION BY MS. HARVIE:

Q. All right. Starting with you, Mr. Penn, would you please provide a brief overview of Ontario Hydro's current nuclear program?

MR. PENN: A. Ontario Hydro's existing nuclear generating stations provide a major part of the province's electricity supply today, and they will continue to do so for the next 25 years.

The overheads which I will be using are all shown in Exhibit 519, and my first overhead is on page 1 of Exhibit 519. It shows the nuclear capacity which will be installed once Darlington nuclear generating station is in-service.

If I may just make a few comments on that overhead on page 1, it shows a pie chart to the left, a pie chart which gives the total system capacity as 32.5 gigawatts by 1993. To the left of the chart is the fossil capacity consisting of coal-fired generating stations and oil which represents 37 per cent of the

system capacity or 11.9 gigawatts. To the right of the pie chart is the nuclear capacity, which represents 43 per cent of the total system or 14.1 gigawatts, and at the bottom is the hydraulic capacity, which represents the balance of 20 per cent or 6.5 gigawatts.

My second overhead on page 2 shows the nuclear capacity that Hydro has assumed in the period 1992 to 2014, which is consistent with the Update, Exhibit 452.

On the vertical axis is shown capacity in gigawatts. On the horizontal axis are the years between 1990 and 2015. The upper curve shows the total capacity of Hydro's generating system over the total period, including all forms of generation. The lower curve shows the capacity which would be provided by nuclear power.

At the end of the period starting 2009 you will notice a drop in the curve, which is when we plan at this point in time to start removing Pickering nuclear generating station A from in service.

My third overhead, which is on page 3, shows the total energy production of our system in terawatt hours. The upper line is the total system per energy production over the total period, and you will note that it rises slightly over that period.

The bottom line shows the nuclear generation production, energy production, over that period, and again shows a decline at the end of the period.

I would like to note that in 1993 when Darlington is in-service nuclear will provide 62 per cent of the energy production to our province and by 2009 it will be about 52 per cent.

Q. What are the main characteristics of Ontario's existing nuclear generating stations?

A. The main characteristics are seven in number and are as follows.

First, the stations are all of the CANDU design in Ontario. CANDU stands for "Canada Deuterium Uranium"; that is, the concept was totally developed in Canada.

The moderator and coolant system that are within the reactors consists of deuterium oxide or heavy water. The fuel is natural uranium; that is, it has natural isotopic composition and is found in nature in Northern Ontario and Saskatchewan, principally in Canada.

The second characteristic is that there is a fully developed infrastructure in Canada to support CANDU stations. The CANDU has a very high

Canadian content. Most of the capital cost and essentially all of the operations and fuel costs are spent in Canada and in particular in Ontario.

[10:25 a.m.]

Third characteristic is that all nuclear stations have high initial capital costs, but low fueling costs so that they are best suited for base load operation. They are the work horses of our system.

Fourth, the CANDUs on the Hydro grid are all integrated four-unit designs. They take advantage of economies of scale associated with large stations and the ability of multi-units stations to use common facilities.

Fifth, the CANDU program has benefited from a fairly high degree of standardization; that is, successive multi-unit stations have repeated design features utilized in earlier stations.

Sixth, for base load operation, CANDUs have the lowest levelized unit energy cost, or what is normally called LUEC, of any supply option available in Ontario, although the economic gap between coal and nuclear has closed in the last few years.

If we were to assume in-service in 2002, as was done in the fossil cost review, a new CANDU

would be expected to have a LUEC of between 10 to 15 per cent less than the lowest cost fossil option, which is a 4 by 800 megawatt conventional steam-cycle plant, utilizing United States supplied coal.

And finally, the seventh characteristic of the CANDU system is that the multi-unit station concept entails a long lead time to build. Typically from the planning stage to first unit in-service is some 12 years. This and the high capital cost reduce planning flexibility of this option.

I would like to mention that my colleagues on this panel will also be addressing important characteristics. Mr. Daly will be discussing the performance of our CANDU systems; Mr. King to my right, the safety of our systems; Mr. Johansen further down, the importance of protecting the environment we enjoy, and lastly, Dr. Whillans will talk about the impact of nuclear energy on our health.

Q. The witness panel will be expanding on many of the points you have raised, but will you also remind the Panel of some of the material available to them on the nuclear option, please.

A. Well, in addition to well over 1,000 interrogatories that this panel and its support staff have answered, Hydro has tabled a number of important

exhibits, containing evidence on the nuclear option. These are listed on page 4 of Exhibit 519 on the next overhead.

They are the Demand/Supply Plan, Exhibit 3; the accompanying environmental analysis, Exhibit 4; an important document known as Ontario Hydro Presentations to the Ontario Nuclear Cost Inquiry, or known colloquially as ONCI; followed by a report to the Minister by an international panel of experts that were convened to review ONCI, and this is given in Exhibit 44.

A further important exhibit is the Ontario Nuclear Safety Review, which contains some four volumes, and is given the title the Safety of Ontario's Nuclear Power Reactors, a Scientific and Technical Review. Those four volumes are Exhibits 184 to 187.

And finally, Exhibit 507, which has the title Materials Relating to the Environmental and Health Effects of Nuclear Generation.

I might also add that there is a substantial amount of technical detail in the seven volume Darlington Safety Report which was provided in response to Interrogatory 9.7.58.

Q. Mr. Penn, will you give the Board a brief description of the CANDU system, please.

A. Yes, I will.

Perhaps I would like to start by commenting that for those that would like more detail than I will supply, that Chapters 3 and 4 of the ONCI document, Exhibit 43, provides that information.

On the next overhead, which is on page 5, is a schematic of the CANDU nuclear power station. And from a straightforward point of view, the CANDU nuclear power station consists of five major components, and if we move from the lower left, clockwise, the first component, of course, is the CANDU reactor itself.

Above that are the steam generators, which from the heat from the reactor raise steam of light water, which then flows into a steam turbine, which is shown at the top of the diagram, towards the centre.

Beyond that is the generator itself which produces the electricity. And the fifth major component is the condenser which takes the low quality steam returned from the steam turbine, condenses it and returns it as feed water to the steam generators.

Having identified the five major components, I would now like to briefly say something about each of them and how they fit together in an integrated fashion.

Again, on the lower left of the diagram is the reactor core. The reactor consists of a metal calandria which contains heavy water moderator. Of course the heat of reactor has biological shielding about it of water and concrete.

Running horizontally throughout the calandria vessel are what are known as calandria tubes, and inside each calandria tube is a pressure tube. The pressure tubes contain the uranium fuel bundles.

The anulus gap is important that exists between the pressure tube and the calandria tube. It serves two purposes, one to insulate the moderator from the hot coolant, and the second is to ensure that if we had a pressure tube failure, it would be detected.

By way of an example, at Darlington there are 480 fuel channels. Each fuel channel contains 13 fuel bundles.

Mr. Daly will be showing you a fuel bundle suitable for Darlington which contains 37 elements. In each element are small cylindrical pellets of uranium dioxide.

Fission of uranium generate heats. It also generates neutrons, which in turn are slowed down in the moderator to cause the chain reaction to continue.

The primary pumps that are shown immediately above the reactor drive heavy water coolant through the reactor and through the steam generators, then back to the reactor.

The primary coolant flow through the stream generators in tubes boils light water on the other side of these steam generator tubes to produce high pressure, high temperature steam, which then flows to the steam turbine, which in turn drives the generator and strips electrons to produce electricity.

You will note immediately below the steam generator and termed the low pressure turbine, there is a condenser, which as I mentioned before, condenses low quality steam that can no longer be used in the steam turbine, and this condensation is caused by water taken from the lake and in turn returned to the lake.

Please note that this circuit is totally independent from other circuits in the power station.

The features which distinguish the CANDU concept from light water reactors commonly employed throughout the world and in particular in the United States, Japan and Europe are as follows: CANDUs use heavy water, D2O, moderator and coolant. The reason is that we also employ natural uranium in our reactors whereas light water reactors that use light water

moderator and coolant employ enriched uranium.

The second important point is that we use horizontal pressure tubes which contain our uranium fuel and in turn facilitates on-power refueling, which light water reactors are not capable of.

Before I leave this diagram, I just want to point to a few things that we will be talking about in more detail in our evidence in chief.

Specifically, we will be introducing to you the term large scale fuel channel replacement, or: LSFCR. That is the process of removing the pressure tubes in the reactor and replacing them with new tubes, a process that we call, in simple terms, retubing.

We will be discussing what causes the end of the service life of these pressure tubes, how we remove them and replace them, and how much it costs to do it and how long it takes to do it.

The second point I would like to bring to your attention, and with respect to the so-called fuel issue at Darlington, is the inlet headers that are immediately above the reactor, and the main coolant pumps that are above those. I point this out because there is a relationship between the nature of those pumps, the size of the headers and the piping that joins the reactor to the issue that is before us at

Darlington today.

I would like to bring to your attention again the steam generators. We do have some issues of concern at our Bruce "A" generating station with steam generators, and we will be discussing with you the performance there, and at our other stations, and what we plan to do about it.

And the final piece of major equipment I want to mention is the generator, again in relationship to the rotor problems at Darlington, which we have now solved.

I might add that Mr. King will be discussing special safety systems, including the containment around the reactors, the special shutdown systems, and the emergency coolant injection systems.

Q. Mr. Penn, would you please describe Ontario Hydro's existing CANDU program.

A. Because of the nature of our multi-unit stations, we operate a nuclear plant at three sites in Ontario. Those are Pickering, Bruce, and Darlington.

The next overhead, which is on page 6 of Exhibit 519, shows an aerial view of Pickering nuclear generating station, which is located on the shore of Lake Ontario, about 30 kilometres east of where we are

today.

The eight reactors are clearly discernible, having cylindrical containment with spherical domes.

We will be referring to Pickering "A" nuclear station which are the four reactors at the lower part of the picture.

The four reactors to the right in the upper part of the picture are Pickering nuclear generating station "B".

You will notice on the lakeshore side of these eight reactors is a rectangular structure which a pressure relief duct which joins to the vacuum building which is the largest building in the nuclear power station, and serves, because it's under negative pressure, like a vacuum cleaner in the event of a major accident in any one of the units.

[10:38 a.m.]

On the land side of the eight reactors you will see two large rectangular buildings. These house the eight steam turbines and generators that provide the electricity from Pickering nuclear generating station.

The next overhead, which is on page 7, provides a view of the Bruce Nuclear Power Development

site. This is located on the shore of Lake Huron and comprises eight reactors, each of about 850 megawatts capacity. In the foreground is Bruce "B" station towards the bottom of the picture. You will notice that compared with Pickering the reactor buildings and containment are square in design.

In the middle of the four reactors is the common services area and the control rooms for this station, and immediately to its left is the large vacuum building that I have previously described for Pickering site. Again on the right is a single, large powerhouse containing the four turbine generators and is about half a kilometre long.

In the middle of the picture you will notice the heavy water production plants, and if you look carefully you will see the distillation towers of the units that remove the heavy water that is naturally in Lake Huron and isolated.

We have at the moment one heavy water plant known as Bruce Heavy Water Plant "B" operating on our site.

To the left of the heavy water plants you will notice a cylindrical reactor building with a spherical dome, and that is the Douglas Point nuclear station which was a prototype of our power reactors and

has now been shut down permanently.

Beyond and at the top of the picture and towards the right in the distance is the Bruce "A" nuclear power station, which is essentially identical to Bruce "B" and of a similar square design.

My last overhead and on figure -- page 8 is a recent aerial view of our Darlington site.

At Darlington there are four reactors, again of a square design similar to the Bruce ones, each of 881 megawatts capacity. All four are currently scheduled to be in-service by March, 1993.

Q. Mr. Penn, you mentioned that Ontario Hydro's nuclear program has benefited from standardization. Would you explain what you mean by "standardization" and describe the benefits that have resulted?

A. The standardization of Ontario Hydro's CANDU nuclear program is founded on several major decisions.

First, Ontario Hydro has refined, together with its consultants and industry, the CANDU concept over more than 35 years.

Second, stations have been built as multiple units, thus reducing the number of sites being employed to a small number, and each station having

four identical units with common service facilities.

The third point to note about standardization is that there are two basic series of designs at Hydro plants: the nominal 500 megawatt units at Pickering and the 850 megawatt units installed at Brucė "A", Bruce "B" and Darlington.

Standardization thus includes fundamental repeats of construction and operation practice. The benefits to design and construction include economy of scale, savings and building on experience.

Less obvious but also important are the benefits to ongoing operations, which include standard and defined operating procedures: less staff training; a multi-unit approach to regulatory licensing; and finally, the generic applicability of update modifications to our units at our stations.

Q. Thank you, Mr. Penn. Turning now to you, Mr. Daly, would you briefly outline the nature of your evidence on nuclear performance?

MR. DALY: A. I would like to cover the following seven areas this morning: first, a brief description of our operating stations, including those under construction; second, some of the performance indicators we use to measure nuclear performance; third, the various nuclear forecasts we produce and the

methodology behind them; fourth, our experience with each of our existing nuclear stations and how we see these stations performing in the future; fifth, some of the improvement programs we have initiated; sixth, our lifetime performance targets and associated programs; and finally, just to give some context, a comparison of our nuclear units with other nuclear units around the world.

Q. Fine. Then turning to your first topic, would you briefly describe the current nuclear generating stations on the Hydro system and the ones that are under construction presently?

A. As shown on page 9 of Exhibit 519 we have four nuclear generating stations fully in-service, each having four units.

A fifth four-unit station, Darlington, is being commissioned with one unit declared in-service to date.

The 17 in-service units are Pickering A, Units 1 to 4, each rated at 515 megawatts; Pickering B, Units 5 to 8, each rated at 516 megawatts; Bruce A, Units 1 to 4, each rated at 769 megawatts, or 848 megawatts if we include the process steam capacity that is sent from Bruce "A" to the heavy water plant to produce heavy water; Bruce "B", Units 5 to 8, each

rated at 860 megawatts; and finally, Darlington Unit 2 rated at 881 megawatts.

THE CHAIRMAN: When you say Unit 2 do you mean two units or one unit there?

MR. DALY: No, I mean just Unit 2.

THE CHAIRMAN: Just one unit?

MR. DALY: Just one unit. Unit 2 was the first unit to be declared in-service.

Page 10 of Exhibit 519 shows the three units of Darlington that are not yet in service, Darlington Units 1, 3 and 4, each rated at 881 megawatts. The scheduled in-service dates for these units are August, '92 for Units 1 and 3, and March, 1993 for Unit 4.

MS. HARVIE: Q. Moving to your second topic, what performance indicators do you use for measuring nuclear performance?

MR. DALY: A. We use a very wide variety of indicators to measure perform, and I will briefly review five of the most frequently used indicators.

These indicators, which are listed on page 11 of Exhibit 519, are the maximum continuous rating, the capacity factor, the capability factor, the incapability factor, and the derating adjusted forced outage rate, sometimes known as DAFOR.

First of all, turning to the maximum continuous rating, abbreviated to MCR. This is the maximum power output level at which a unit is expected to be able to operate continuously. Operation at less than this level is considered to be a derating.

In general, the MCR is defined as the maximum power level the unit was originally designed to achieve. Once a unit is in commercial operation, however, the MCR and hence the overall energy output can be increased through a combination of in-service design modifications and operational improvements, and to date we have increased the MCR on 12 of our 17 nuclear units through such improvements.

Second, capacity factor. Capacity factor is a measure of a nuclear unit's actual energy production, and the formula for calculating capacity factor is shown on page 12 of Exhibit 519. Capacity factor is defined as the percentage of a nuclear unit's perfect energy output which is actually produced in a specified time period.

Perfect energy production is the energy that would be produced by operating a unit continuously through a period with no outages or deratings. We calculate perfect energy production by taking the unit's maximum continuous rating and multiplying it by

the time period.

In practice the capacity factor of a nuclear unit may be limited by both internal and external constraints. An internal constraint, for example, would be an equipment problem. On the other hand, a transmission line limitation or a lack of demand would be an example of an external constraint. The capacity factor incorporates all production constraints for any reason whatsoever, both internal and external.

Moving on now to the third index, capability factor, this is a measure of a nuclear unit's potential energy production, and the formula for calculating capability factor is shown on page 13 of Exhibit 519.

Unlike capacity factor, capability factor considers only production constraints that are internal to a unit. Capability factor is therefore a better measure of unit performance since transmission limitations and load constraints are not included.

The capability factor is defined as the percentage of perfect energy production which is available in a specified time period.

Since the capability factor of a nuclear unit is a measure of potential production the

capability factor is always greater than or equal to the capacity factor. The difference between the capability factor and the capacity factor is the energy that could have been produced but wasn't due to some external factors. In practice, the differences are pretty small.

However, in the late 1970s and early '80s inadequate transmission capacity locked in a significant amount of energy at our Bruce site and resulted in capacity factors that were up to 8 per cent lower than our capability factors.

The fourth index, incapability factor, which is also shown on this page, is a related measure to capability factor. Incapability factor is the percentage of perfect production which is not available due to internal causes.

Capability factor and incapability factor usually sum to 100 per cent.

Incapability factor is commonly used to identify reasons for lost production. For example, we track the lost production due to major nuclear unit systems and equipment, as I will describe later on, and we often use incapability factor as one of the measures of equipment performance.

Finally, the fifth index, the derating

adjusted forced outage rate, or DAFOR. This is shown on page 14 of Exhibit 519. This is formally defined as the total energy production lost through forced outages, forced extensions to schedule outages, and forced derated operation as a percentage of the scheduled energy production.

Scheduled energy production in this instance is the perfect energy production minus scheduled production losses, such as planned and maintenance outages.

Some forced outages and deratings are inevitable due to periodic equipment breakdown or wear-out. However, it is preferable to keep the DAFORs as low as possible. An unscheduled production loss is in general more expensive than a scheduled production loss, such as a plant outage, and such unscheduled production losses may result in equipment or system stresses.

Q. Mr. Daly, once Ontario Hydro measures these performance indices and comes up with a result are there standards that the results can be compared with?

[10:55 a.m.]

A. Yes, there are. Our standards for capability factors and derating adjusted forced outage

rate are based on average values from 163 pressurized water reactors around the world as reported annually by the International Atomic Energy Agency. We chose pressurized water reactors as an external reference because they represent the best of the non-CANDU types.

I think it is important to note that a standard is a type of benchmark by which we can compare ourselves to the rest of the world.

Internally, however, we set performance targets which generally surpass world standards. We aim to achieve performance targets for capability factors which are higher than world standards except, of course, when this would be unrealistic, for example, when a unit is down for a major planned outage such as retubing.

Page 15 of Exhibit 519 shows the 1991 performance standards and the corresponding 1991 results for the Pickering and Bruce generating stations. As you can see the "B" stations bettered their standards for capability factor and DAFOR, for DAFOR our low values are better, whereas the "A" stations did not achieve the standards. I will be dealing with this in more detail later.

Darlington is not included in this chart due to the operational problems encountered with Unit 2

which I will describe later.

Q. The third topic you wanted to cover was forecasting. You have described how actual performance is measured, but what about forecast performance, could you describe, please, how you forecast nuclear performance?

A. We produce a number of nuclear forecasts incorporating short-term, mid-term and long-term nuclear performance projections.

First the short-term. The short-term nuclear forecast is primarily focused on nuclear capacity and energy production over the near term winter periods. These forecasts are used for short-term strategic planning purposes such as system reserve planning.

The mid-term nuclear forecast is updated and issued several times a year. This contains generation performance forecasts for the current year and five years into the future. We use the mid-term nuclear forecast for rate setting purposes, budgets, and the front end of business plans. It contains information that sets operational energy planning and fuel purchasing.

Accurate forecasting is, of course, very important. We use a number of techniques including

outage frequency and duration analyses, actual versus forecast variance analyses, and extrapolation of past trends to check the validity of our forecasts.

In recent years we have found ourselves overestimating the production from our older "A" stations and we have made a number of downward adjustments in our "A" station forecasts based on this experience.

We are also developing a probabilistic forecast model to assist in predicting nuclear unit performance. This model simulates the performance of 18 of the key systems, and from this simulation, projects the performance of the unit as a whole.

The model is still in the development stage with a preliminary version scheduled to be completed this year. And when we have that fully developed, the model will provide another useful check on the baseline nuclear forecast.

In addition to the short-term and the mid-term forecast that I have just described, a long-term nuclear forecast is issued annually. The long-term forecast covers the period from the current year right up to the end of the life of all the nuclear units. These forecasts are used for long-term strategic planning purposes such as the Demand/Supply

Plan and the Plan Update.

Q. Since it's the long-term forecast that's used in long-range planning, would you please describe that in more detail.

A. The long-term forecast of nuclear capability factor is calculated based on a forecast of planned outages, unplanned outages and deratings.

The frequency and duration of planned outages are forecasts based on required inspection and maintenance schedules for major equipment, regulatory inspection guidelines and resource availability.

Unplanned outages and deratings are projected by considering such things as past performance of a unit, age, vintage, engineering judgment as to the future reliability of major systems, specific upgrading plans and staff and resource levels.

The specifics of our forecast methodology are discussed in more detail in Interrogatory 9.2.88.

A long-term forecast is produced by looking at two separate time periods. First, the business planning time frame which extends 10 years into the future, and second, the years beyond the business planning period, i.e., the period beyond 2001.

As you might imagine, a long-term forecast is fairly detailed over the business planning

period, that is over the next 10 years. Information regarding specific outage and upgrading plans, as well as the available resource levels are well specified.

The performance forecast over these 10 years is therefore quite detailed and includes year-by-year projections of capability factors.

For the years beyond the business planning time frame, i.e., beyond 2001, considerably less detailed information is available. Therefore, the long-term forecast over this period is less specific.

Generally, this part of the forecast is composed of projecting an average capability factor over the entire time frame outside of retubing outages.

It is important to note that the long-term forecast may, depending on the station, extend up to 40 years into the future. The difficulty in producing such a long-term projection is that there is little actual experience available with nuclear units of this age. We ourselves have no nuclear units over 21 years old, and there are virtually no comparable nuclear units around the world over 25 years in age.

So although we examine the many variables I have listed, in the end the long-term forecasts come down to reasoned engineering judgment, especially when

considering the years beyond the business planning period.

For this reason we do not attempt to produce one single number that is representative of our long-term performance forecast. Instead, we forecast a range of values so that we can capture the performance uncertainties both positive and negative into the forecast.

In practice we expect that our actual performance will fall within the forecast range with an 80 per cent probability.

Q. Mr. Daly, after you establish this range of values, how do go about selecting operating targets and forecast numbers?

A. As I discussed earlier, we generally set ourselves challenging but achievable performance targets that surpass world performance standards.

So in general, our targets are at the upper end of our forecast performance range. In other words, we set ourselves targets that have less of a probability of being met than our median or most probable forecasts.

Our performance target for our "A" stations is to achieve an average capability factor of 85 per cent after retubing and rehabilitation. For our

"B" stations our aim is to achieve average capability factors of 85 per cent both before and after retubing. These target values are some 5 to 10 per cent higher than our median forecasts.

Actual unit performance will, of course, vary from year to year, due to variations in planned outage schedules and the like. In other words, some units may perform better than forecast some years, worse than forecast other years.

Over the long haul what we are trying to project is the overall average performance over the entire long-term forecast period.

Although we certainly prefer to forecast performance ranges as I have just described, it's often necessary to pick one value that represents long-term performance for use in planning studies like the Demand/Supply Plan Update. In general, the median values used in the Demand/Supply Plan Update are 5 to 10 per cent lower than our targeted values. And I will be discussing the specific values used in the Update a little later.

Q. Let's move on to your fourth topic, individual station performance. You have described for us the kind of performance indices you used to measure nuclear performance, and you have given some insight

into the forecast process.

I would like to ask you to describe actual and forecast performance for your nuclear units beginning with your older station Pickering "A" and tell us what kind of performance you see in the future.

A. Page 16 of Exhibit 519 shows the actual annual capability factors of Pickering "A" from 1971 through to the end of 1991 on the left-hand side of the chart, and the forecast capability factors consistent with the Demand/Supply Plan Update to the end of 2014, and these are on the right-hand side of the chart.

After an initial teething period which ended in the mid-70s, Pickering "A" performance was excellent with capability factors exceeding 80 per cent until 1982.

The unplanned concurrent retubings of Pickering Units 1 and 2 kept capability factors low throughout the mid-80s.

Retubing of the Pickering "A" units had always been anticipated, however retubing of those units was brought forward several years following the 1983 failure of a Unit 2 pressure tube. At the time Unit 2 came down, this type of pressure tube failure mechanism was unexpected.

Q. Mr. Daly, would you please describe that 1993 failure of the Unit 2 pressure tube at Pickering "A" in a little more detail?

A. Certainly.

Page 17 of Exhibit 519 is a schematic diagram of a typical fuel channel, and Mr. Penn referred to this briefly earlier on.

As you can see, a fuel channel is composed of two concentric tubes, an outer tube called the calandria tube and an inner tube called the pressure tube.

The fuel is contained within the pressure tube and it is through the pressure tube that the heat transport water flows via the feeder pipes, removing the heat produced by the fuel.

The pressure tube and calandria tube are separated by spacers or garter springs as they are often called. These garter springs are necessary since the pressure tube normally sags due to the weight of the fuel bundles, and would otherwise touch the calandria tube.

The space between the pressure tube and calandria tubes is filled with a gas also described by Mr. Penn called the anulus gas. And as he mentioned, one of the main functions of the anulus gas is to

provide thermal insulation between the pressure tube which is hot and the calandria tube which is cold.

Turning to the failure on Unit 2, this was due to the pressure tube contacting the surrounding calandria tube and hence creating a cold spot on the pressure tube. This cold spot tended to concentrate the deuterium and hydrogen which were present in the pressure tube and this eventually leads to the formation of what we call blisters. These blisters became weak points in the wall of the pressure tube which lead to the failure of the tube in August of 1983.

The cold spots in Units 1 and 2 are the result of an insufficient number of garter springs. The Pickering "A" Units as well as Bruce Units 1 and 2, have only two springs per fuel channel. Two springs were not sufficient to maintain separation between the pressure and calandria tubes over a long period of time. In addition, many of the garter springs have moved out of position and could not adequately separate the pressure and calandria tubes. The result was that over the years several of the pressure tubes sagged, enough to contact the calandria tubes.

All designs following Bruce 2 use four springs as shown in this figure to avoid this problem.

Q. Can we go back to page 16 of Exhibit 519, Mr. Daly, and would you tell the Board what impact this failure on Unit 2 had on Pickering "A" performance?

A. As a result of the failure on Unit 2, both Units 1 and 2 required immediate retubing. While the retubing of Units 3 and 4, which are marked on the chart, were moved forward several years.

In addition, increased pressure tube inspection and hence planned outage time was required at a number of other stations.

Due to this accelerated retubing schedule and the need to retube Units 1 and 2 in parallel, the retubing outages were very long. Approximately five years to retube Unit 2, and 4 years to retube Unit 1.

The retubing of Unit 3 was completed in much less time, about two years, due to better planning and the experience we had gained from Units 1 and 2.

The retubing of Unit 4, which is now in progress, is expected to be complete in about 19 months.

So in summary, on Pickering "A", the station is currently about 80 per cent complete in its program of reactor retubing and rehabilitation. Pickering Units 1, 2 and 3 have been retubed and

rehabilitated and they have an average post-retubing capability factor to date of 75 per cent.

Currently through several improvement programs that I will describe later, we are targeting to achieve an 85 per cent capability factor for all units after retubing and rehabilitation is complete.

As I said earlier, despite our preference for forecast ranges, the Demand/Supply Plan Update required us to use single point estimates for long-term nuclear performance.

And the forecast performance used in the Demand/Supply Plan Update as shown on page 16 here, has station capability factors at the 75 per cent level in the post-2001 period.

You might also note that the forecast line does not extend through to the end of the year 2014, and this is because the station is to be fully retired by the year 2013. In fact, we plan to begin sequentially removing the units from service starting with Unit 1 in 2011 and the last unit, Unit 4, would be retired in 2013.

Q. Turning to your other "A" station could you outline your actual experience and forecast performance that you expect from the Bruce "A".

A. Moving now to page 18 of Exhibit 519,

this shows the actual capability factors of Bruce "A" from 1977 through to the end of 1991 on the left part of the chart, and the forecast performance on the right-hand side of the chart, consistent with the Demand/Supply Plan Update to the year 2014.

A few early problems in the days just after start up keep capability factors slightly below 80 per cent in the late 70s. Performance at Bruce "A" however dramatically improved through the early 80s with several of the units being classed as top world performers in this period. From 1985 onwards, however, performance steadily declined due to several factors.

The first of these factors, the amount of fuel channel-related work increased significantly after 1985. Part of this workload, for example, pressure tube shifting to accommodate dimensional changes was expected at the time the station started up. However, as a result of the pressure tube failure on Pickering Unit 2, there was a large increase in inspection and remedial maintenance outage time.

The second factor, in the years since 1988 seem generator-related deratings and boiler tube failures on Units 1 and 2 have resulted in significant loss of production.

Finally, maintenance budgets and

maintenance staff levels over the mid to late 80s were inadequate to support both the increased pressure tube work and steam generator work, as well as cope with routine preventative maintenance activities.

The outage time needed to cope with the increased work program, as well as growing backlog of preventative maintenance work resulted in capability factors falling over the 1985 to 1990 period.

Bruce "A" is now entering a period of retubing and intensive rehabilitation and preparations are now being made for the first unit to be retubed and rehabilitated starting in 1994.

[11:12 a.m.]

This will be followed by a second retubing and rehabilitation starting in 1997. Rehabilitation work that we can perform outside of retubing outages will be performed on Units 3 and 4 over the next eight years.

The Bruce "A" station is expected to operate at capability factors in the 50 per cent range for the remainder of the 1990s due to the scope of this work program. Several other improvement programs that I will describe later are under way to improve the operating performance of the Bruce "A" units.

The target capability factor for all

Bruce "A" units is 85 per cent after this retubing and rehabilitation work is complete. This level of performance is targeted because of the significant scope of the rehabilitation work and the anticipated effectiveness of other performance improvement initiatives.

Just like Pickering "A", the Demand/Supply Plan Update assumes a 75 per cent post-retubing performance level as shown on page 18 for all Bruce "A" units in the post-2001 period.

Q. Mr. Daly, would you turn now to the "B" stations and summarize the actual and forecast performance of, say, Pickering "B"?

A. Page 19 of Exhibit 519 shows the actual annual capability factors of Pickering "B" from 1983 through to the end of 1991 and the forecast performance consistent with the Demand/Supply Plan Update to the year 2014.

Pickering "B" performance has been excellent to date with the station averaging capability factors greater than 80 per cent for all years except for 1990 when the station was shut down for a routine inspection of a vacuum building.

In 1991 three of the Pickering "B" units operated at capability factors exceeding 90 per cent,

including two units with capability factors of 99 per cent. The average capability factor of Pickering "B" from in-service to the end of 1991 is 85 per cent. We are targeting to maintain this 85 per cent level over the long term except for retubing and SLAR outages.

SLAR is an acronym that stands for Spacer Location and Repositioning. SLAR is a tool that moves the garter springs that I have previously described. It moves those springs in order to ensure separation between the pressure and calandria tubes. Pickering Units 5 and 6 are the only "B" units that require SLAR work, and those SLAR outages are planned for these units in 1997 and 1998.

For Pickering "B" the Demand/Supply Plan Update assumes a long-term capability factor of just above 80 per cent as shown on the overhead, except for retubing and SLAR outages. Retubing of these units will begin in 2010.

Q. All right. And what about Bruce "B"?

A. Page 20 of Exhibit 519 shows the historical annual capability factors of Bruce "B" from 1984 through to the end of 1991 and the forecast performance consistent with the Demand/Supply Plan Update to the year 2014.

Just like Pickering "B", the performance

of Bruce "B" to date has been excellent. In 1991 three of the Bruce "B" units achieved capability factors of 90 per cent or greater. Overall, the station average capability factor from in-service to the end of 1991 was 89 per cent.

We are targeting to maintain at least an 85 per cent capability factor over the long term except for retubing for the Bruce "B" units.

Similarly to Pickering "B", the Demand/Supply Plan Update assumed a long-term capability factor of just above 80 per cent except for retubing outages for Bruce "B". Retubing of the Bruce "B" units is expected to begin in 2011.

Q. Mr. Daly, I would like to explore the differences between the "A" and the "B" stations in more detail later, but would you move on, please, to Darlington?

There has been quite a lot of media coverage of two significant technical difficulties at your newest station, Darlington. Would you outline, please, what those problems are.

A. There have been two significant problems at Darlington discovered during the commissioning of Units 1 and 2.

One of these problems is on the

conventional side of the plant and is associated with the generator rotor. The second problem is on the nuclear side of the plant and is associated with the fuel bundles.

Taking the first problem, the cracking of the generator rotors, this was discovered during the commissioning of Unit 2 in early 1990. The original generator rotor had to be scrapped and a modified rotor was installed in Unit 2 prior to startup. In February of 1991 a second crack was discovered on the Unit 1 rotor. Further modifications to the rotor design have since been made.

The second problem was some fuel bundle damage that was discovered in November of 1990 during a routine refueling operation on unit 2. Fueling could not be carried out and the unit was shut down in December of 1990.

Fuel inspections revealed that some of the fuel bundles had been damaged whilst they were in the reactor. Similar damage has been found on Unit 1 in early '92. Work is currently under way on resolving this problem.

Q. Could you elaborate further on the generator rotor work at Darlington?

A. Page 21 of Exhibit 519 is a diagram

of a Darlington generator rotor. The rotor of course is the rotating part of the generator. The upper part of this diagram shows about half of the full rotor. In the lower part of the diagram is shown a cutaway detail of the right-hand side of the rotor.

As stated in our response to Interrogatory 9.7.287, the original generator rotors were prone to cracking in two different locations: here at the lead-in studs marked as "A" on the diagram, and secondly at "B" on the diagram where the wedges hold the electrical leads in place.

In general, the cracking was the result of bending stresses caused by flexibility of the generator rotor shaft. These generator rotors in Units 1 and 2 have been extensively modified to resolve the problem. The work has been performed under warranty and at the manufacturer's expense.

All Darlington units will be outfitted with modified rotors before being declared in-service or, in the case of Unit 2, before being returned to service.

We have only had a short period of time with the current rotor modification on Unit 1. However, no problems were evident. Further in-service delays due to the generator rotor problems are not

anticipated at this time.

Q. Could you provide an update on the fuel bundle damage?

A. What I am holding now is a fuel bundle of a type used at our Darlington and Bruce stations. A drawing of this type of bundle is shown on page 22 of Exhibit 519.

Each bundle is composed of 37 fuel elements which contain the uranium oxide pellets. The elements are welded to two end plates, one at each end, to form a bundle. There is no new or used uranium in this bundle.

Inspections indicated that some fuel bundles had cracked end plates, in this area (indicating) around the welds on the end plate, and they also had excessive bearing pad wear in this area here (indicating), near the top of the bundle where the fuel bundle rests on the pressure tube.

We did a lot of extensive tests and inspections. Particularly, in July of 1991 a number of tests were performed on reactor No. 2, which remains shut down, to simulate reactor conditions.

The results of those tests indicated that the end plate damage was due to pressure pulses in the heat transport system.

Page 23 of Exhibit 519 is a simplified diagram of a typical CANDU reactor. So if we follow the heat transport water it follows a path from the heat transport pumps, through the feeder pipes, passing through the inlet header. Let's just go back to the inlet header.

So it passes through the inlet header, through the feeder pipes, to the pressure tubes, up to the steam generators and back to the pumps.

What appears to be happening and causing the damage on the end plates and the wear on the bearing pads is as follows.

Each of the heat transport pump impellers has five vanes. This means that five pulses of heat transport water are pumped for every complete turn of the heat transport pump shaft. Since the heat transport pumps revolve at 30 revolutions per second, so we get five times 30 or 150 pulses of water are being pumped every second.

It so happens that those pressure pulses occurring at 150 times a second are amplified by the particular piping configuration and particular lengths and diameters of piping that we use in the heat transport system at Darlington, and the result of that amplification of the pressure pulses is that this fuel

string, which is composed of 13 of these bundles lying end to end, this rocks back and forth at about 150 times per second. This back and forth motion of the fuel bundles is straining the end plates and wearing down the bearing pads.

We have done a tremendous amount of work to determine the cause of the damage and come up with possible repair strategies. Several possible repair strategies have been proposed.

The proposed solution to the fuel damage problem consists of replacing the current five-vane heat transport pump impellers with seven-vane impellers. The use of seven-vane impellers will change the frequency of the 150 cycles per second pressure pulses so that they should not be amplified by the piping.

These new pump impellers have been ordered and installation of the first set of impellers is to begin in May of '92 on Unit 3 during commissioning of that unit. We chose Unit 3 for the first impeller installation since this will allow the work to be done under non-radioactive conditions.

If the seven-vane impeller modification proves to be successful, then we will install it in all other units.

The use of these new impellers as a fix to the fuel damage problem was based on a computer simulation of the hydraulics of the heat transport system, and there is some uncertainty as to whether or not this repair will be totally adequate when installed in the plant on an operating unit. Further testing of the impellers will be required after they are installed on Unit 3 in order to gain that actual experience.

If the impeller changeout is not successful, then an additional or an alternative solution which involves modification to the heat transport system piping will be required. Preparatory work on the piping modification has been started so as to avoid too long a delay if it turns out to be necessary.

If the piping modifications are required, then further in-service delays of about six months per unit will be required.

Would the Panel care to have a look at the fuel bundle?

DR. CONNELL: Sure. The pressure tube just fits right around it?

MR. DALY: Fits right around there.

DR. CONNELL: And this is full of...?

MR. DALY: Heavy water, yes.

MS. HARVIE: Perhaps after the Panel has had an opportunity to look at it we will leave the fuel bundle here so parties can inspect it over the break if they wish.

There is a photograph of it or a replica of it in the overhead package.

MR. GREENSPOON: It should be an exhibit, Mr. Chairman.

THE CHAIRMAN: Technically, you are right, Mr. Greenspoon. Do you want it as an exhibit?

MR. GREENSPOON: I do.

THE CHAIRMAN: We can mark it as an exhibit, if you want to do that. It has been referred to. There is no question you are technically correct.

THE REGISTRAR: Number 524.

---EXHIBIT NO. 524: Fuel bundle.

THE CHAIRMAN: I think with that interlude we will just take a break now for 15 minutes, if that's all right, Ms. Harvie?

MS. HARVIE: Yes.

THE REGISTRAR: Please come to order. This hearing will recess for 15 minutes.

---Recess at 11:28 a.m.

---On resuming at 11:45 a.m.

THE REGISTRAR: Please come to order.

This hearing is again in session. Be seated, please.

THE CHAIRMAN: Two administrative matters, we are stopping for the noon break at 12:30, and continuing at 2:30, and we will stop for the day no later than 4:45.

Ms. Harvie?

MS. HARVIE: Q. Mr. Daly, what kind of performance do you expect from Darlington once all the units are fully operating?

MR. DALY: A. Just as with Pickering "B" and Bruce "B", we are aiming to achieve a capability factor target of 85 per cent except for retubing once the current problems are resolved.

However, in the Demand/Supply Plan Update a value of 80 per cent for the forecast long-term capability factor except for retubing was used for planning purposes.

Q. Mr. Daly, I would like to step back a bit and get a larger view of nuclear performance.

Would you please identify on a system-wide basis some of the equipment problems that you have been experiencing?

A. Certainly.

Page 24 of Exhibit 519 shows the incapability from 1987 to the end of 1991, attributable

to the major systems on the "A" and "B" stations.

The major systems are shown listed down the left-hand side of the chart. The right-hand side of the chart shows how much incapability in per cent was attributable to each of the major systems.

The systems are listed starting with the highest contributor to system incapability at the top, and moving down to systems which contribute less incapability.

Of most concern, as you can see from the figure, are fuel channels which includes pressure tubes and steam generators. These two contributors have been responsible for much of the incapability over the last five years.

In addition, a large component of the incapability due to these two systems has been forced. The black section of each bar or the left-hand side part of each bar on the chart represents the forced incapability.

THE CHAIRMAN: Just a moment. I would just like to mention that in the photocopies that have we have got at least, you can't distinguish between the dark and the light, and I don't know how you want to rectify that, but if that's of any significance...

MS. HARVIE: Perhaps I can give you my

copy, Mr. Chairman.

THE CHAIRMAN: I can see it on the screen right now, I am just thinking of the long-term.

MS. HARVIE: What we will do then is replicate this page and produce new copies for people.

THE CHAIRMAN: Thank you.

MR. DALY: In the last of the five years, that is in 1991 only, the fuel channels and the steam generators accounted for about 12 per cent of the total incapability. This is not actually shown on the chart.

They contributed about 12 per cent in 1991 only, about 6 per cent due to each.

So, while the incapability due to fuel channels is declining with time, the incapability due to steam generators is increasing with time.

Several initiatives are under way to reduce the overall amount of incapability due to these systems and hence improve station performance.

MS. HARVIE: Q. In order to understand the pressure tube situation a little better, what are the expectations for pressure tube outage durations and pressure tube lives in the existing stations?

MR. DALY: A. Retubing is now being performed on a planned, as opposed to an unplanned, basis. This means that the necessary materials and

resources are now in place well before retubing is to begin.

In addition, the retubing experience gained so far has resulted in improved planning and resource allocation with the result that retubing outage durations continue to fall. All those factors serve to lessen the amount of incapability likely to be incurred in the future.

Mr. Penn will be discussing this in more detail in his evidence.

Pressure tube lives for the "A" stations vary from unit to unit, but they are all less than 25 years for units with their original pressure tubes.

The retubed "A" units have pressure tube lives of 30 years.

Pressure tubes in all our "B" station and Darlington are expected to be fit for service for at least 30 years.

Constraints on the life of pressure tubes have been related to contact with the calandria tube that we have previously discussed, inadequate elongation allowances, manufacturing flaws, and excessive deuterium ingress into the tube.

Almost all the "A" stations have been affected, or almost all the "A" station units have been

affected by one of more of these problems to date. However, a number of measures have been developed to extend the lives of the pressure tubes of these stations, for example, the SLAR mechanism we discussed previously moving the garter springs to the design locations to maintain this separation between the pressure tube and the calandria tube.

Improvements have also been made at the "B" stations and Darlington in order to achieve the full 30-year life of all pressure tubes.

First, the number of garter springs was increased to four, and their design was improved to make them less prone to move out of position.

Secondly, manufacturing inspection methods have been improved over the years and design modifications have been made to allow for increased elongation.

Thirdly, improvements were made to the overall fuel channel design in a way the deuterium ingress into the pressure tubes will be reduced.

Finally, all the "A" stations and "B" stations as well as Darlington have implemented routine pressure tube inspections under the in-service inspection program and the periodic inspection program which are regulated by the Atomic Energy Control Board.

These inspections ensure that tubes are fit for service, and they also give us advance warning of any developing problems. This in turn makes it easily to plan major outages.

Q. What about the other major cause of incapability, the steam generators?

A. Incapability due to steam generators has increased significantly in the last few years. Much of the lost electrical production attributable to steam generators has been at the Bruce "A" station. The steam generator problems at Bruce "A" are due to hard deposit buildups inside the steam generator, and to tube leakage due to vibration and corrosion.

Bruce Unit 2 is currently shut down due to boiler tube leakage as a result of corrosion, and we expect that this unit will remain out-of-service for most of 1992.

Several repair options are being considered including a contingency plan that calls for replacement of one or more of the steam generators during a later retubing outage.

We are taking a number of actions to reduce the production losses associated with steam generators. These actions involve better inspections, better maintenance, and better chemistry control.

First, we have scheduled major boiler tube inspection campaigns at a number of our stations, and in order to support an inspection schedule of that scope we have invested in the latest state-of-the-art boiler tube inspection equipment. These inspections give us the opportunity to perform maintenance in a more proactive and cost-effective manner.

Second, we have developed and are continuing to develop a number of remedial maintenance techniques. For example, mechanical steam generator cleaning procedures were developed and have been used at Bruce Units 1 and 2 over the last three years to successfully remove hard deposit build up.

In addition, our investment in boiler tube plugging equipment has reduced and will continue to reduce outage time.

Third, chemical control of our feed water at all our stations has been improved in order to reduce the steam generator deposit build up.

Also, the steam generator chemical cleaning technique, which is now being developed for Bruce "A", will benefit all other stations requiring this procedure in the future.

The aim of all this effort is to manage and eventually reduce the amount of incapability

attributable to steam generators. As part of this process we are attempting to minimize the amount of unplanned outage time by increasing the amount of planned outage time for those inspections and remedial maintenance.

Q. Mr. Daly, your fifth topic was performance improvement programs. What plans do you have to improve equipment performance, or for that matter, overall nuclear performance?

A. We have launched a number of initiatives in the past few years in order to improve the performance in our "A" stations, our older stations, and to maintain the excellent performance of the "B" stations.

These programs are aimed at making improvements in either the equipment performance, resource levels, or management controls. I would like to summarize for you some of the main programs that we now have in place.

First, there is a technical assessment program, this is aimed at improving equipment performance by assessing the type of maintenance needed at each nuclear station over its remaining life. An example of this program is the Bruce "A" rehabilitation program. These technical assessments examined on a

system by system basis projected failures of equipment throughout of the life of the plant and identified those measures that are necessary to maximize reliability.

From this information the annual costs required to maintain high plant performance over the lifetime can be projected.

We have described this Bruce "A" rehabilitation program in our response to Interrogatory 9.2.131.

The second of the improvement initiatives I would like to describe is the nuclear hiring program. This program was started in 1988 and is aimed at improving resource levels at our nuclear stations. The program is aimed at getting sufficient staff to facilitate improved performance by the mid 90s, retube Pickering Units 3 and 4 on schedule, and commission Darlington Units 3 and 4 on schedule.

In addition to the hiring program, the nuclear higher program, operations, maintenance and administration expenditures have been increased over the past three years or so after a period of restraint in the mid-80s, in order to ensure that maintenance resources are adequately funded. These expenditures will serve to reduce the backlog of maintenance

activities at our stations.

Third in the area of management controls, the quality improvement process was introduced in 1990, and that's aimed at achieving and sustaining excellence in the operation, maintenance and support of all our nuclear generating stations.

The process is designed to develop and implement a number of individual processes to improve the quality operations in the fields of safety, reliability and cost.

We described this as a process, a quality improvement process rather than a program because the intent is to create and maintain an environment of continuous improvement.

Some of the specific production related processes include root cause, determination and correction, maintenance backlog reduction, improved control of modifications, improved documentation, and improved outage and resource planning.

These major improvement programs currently in place are described in our response to Interrogatories 9.44.32 and 9.2.122.

Q. Mr. Daly, the sixth area you were to cover was lifetime performance. What are your targets for overall nuclear performance over the lives of your

existing stations?

A. The target lifetime capability factor for both Pickering "A" and Bruce "A" is 75 per cent. The target lifetime capability factors for these stations was originally 80 per cent, but was reduced last year in light of our experience to date and our expectations for the near term.

The Ontario Energy Board also recommended this value in its 1990 report.

The target lifetime capability factors for Pickering "B", Bruce "B" and Darlington are 80 per cent, and all these lifetime figures are over an assumed 40-year life.

Q. Why are your forecasts for the "B" stations higher than for the "A" stations?

A. Page 25 of Exhibit 519 outlines the cumulative capability factor of the "A" and "B" stations to date.

This figure shows that for every reactor year since in-service, the "B" stations lifetime average performance has consistently outperformed the "A" stations.

After 58 unit years of cumulative experience, the average lifetime capability factor of the "B" stations was 87 per cent, whereas the

corresponding figure for the "A" stations was 80 per cent, 7 per cent improvement.

The better performance of the "B" stations is due to a number of improvements which were made in design, construction, commissioning, operation and maintenance as a result of learning from our experience on our older "A" stations.

In design, for example, improvements were made in the design of the fuel channels for the "B" stations as a result of our "A" station experience. Hence, retubing of the "B" stations is projected to occur much later in the life of the station.

In addition, the amount of fuel channel remedial maintenance prior to retubing is also expected to be much reduced on the "B" stations.

In construction, pressure tube installation techniques were much improved by the time the "B" stations were being built, as a result of installation problems experienced and resolved at Pickering "A" and Bruce "A" .

During commissioning the more rigorous quality assurance programs were implemented during the commissioning of the "B" stations which provide greater assurance of avoiding problems later on.

During operations the experience gained

on the "A" stations has been transferred to the "B" stations on a continuous basis to promote improved performance. For example, our retubing experience on Units 1 and 2 at Pickering "A" has already resulted in shorter retubing times for Pickering Units 3 and 4. This retubing experience will be transferred to the "B" stations in order to reduce retubing times even further.

As another example, experience with steam generator deposit build up at Bruce "A" is leading to tighter control of feed water chemistry at all our stations to minimize the problem.

[12:04 p.m.]

In addition, the techniques that are being developed at Bruce "A" to remove the deposits such as mechanical and chemical cleaning techniques are expected to benefit all stations with steam generators of a similar design.

Finally, the programs I have already discussed, such as the nuclear hiring program and the quality improvement process, are being implemented very early in the lives of the "B" stations. By implementing these programs at the "B" stations now while the units are young the programs are expected to have a much longer payback period. Our expectation

overall is that the "B" stations will continue to outperform the "A" stations over their 40-year life.

Q. Is there any formal program in place to ensure that your nuclear stations do achieve their full 40-year lives?

A. Yes, there is. We are implementing what we call a nuclear plant life assurance program.

This program is primarily designed to ensure the full plant life of 40 years is economically achieved by ensuring that all the critical components achieve their respective design lives.

As a secondary objective in this program the nuclear plant life assurance program is also designed to maintain the option of extending plant life beyond 40 years. However, there are no plans for plant life extension in place at this time.

The nuclear plant life assurance program works by first identifying all the critical components; that is, components that could limit the overall life of the station.

The second step is to review all component degradation mechanisms and assess the condition and remaining life of the critical components.

Finally, new inspection, maintenance and

operating procedures for each component are defined and implemented as necessary.

We began this nuclear plant life assurance program in 1988. Currently, we are in the process of completing the condition and remaining life assessment of all critical components at all nuclear stations. By the end of this year recommendations and actions arising from these assessments will be issued and implementation may then begin.

Our responses to Interrogatories 9.2.55 and 9.9.37 covers this in more detail.

Also, the Pickering "A" retubing and rehabilitation program, which is nearing completion, and the similar Bruce "A" program of retubing and rehabilitation also contribute greatly to nuclear plant life assurance objectives.

Q. Mr. Daly, you have outlined for us how you measure nuclear performance and how you expect performance to change in the future. You have also talked about setting performance targets, and you have given us some insight into the programs you have developed to meet those targets.

The final topic you wished to cover dealt with external comparisons. How do your CANDU units compare to the rest of the world?

A. We use capacity factor data collected from about 300 of the world's nuclear units rated at over 500 megawatts. Although capability factor data would theoretically be more suitable, it is not used for unit-by-unit comparisons because of some shortcomings in availability and consistency of the information. Capacity factor data is more readily available and less open to different interpretations.

Q. All right. Well, how have your units performed in comparison to the rest of the world in terms of annual capacity performance?

A. Page 26 of Exhibit 519 shows the annual average capacity factors since 1973 for all Ontario Hydro units. It also shows the annual world averages for pressurized water reactors and boiling water reactors. These figures are based on production since the date of first electrical production so they include Darlington.

It can being seen that our early performance was well above the world average for the other types but the performance deteriorated following the emergence in 1983 of pressure tube problems.

Meanwhile, performance of the other reactor types has progressively improved. In 1990 Ontario Hydro's annual average of 62.4 per cent was

below both of the other two types, but our average improved to 66.4 per cent in 1991.

In 1991 we had two units in the top 10 in the world, including the second placed unit, Pickering 6.

Q. And what have been the world average capacity factors over the elapsed lives of plants for each of the major types of reactor?

A. In terms of lifetime average capacity factors to the end of 1991 weighted by unit sizes and lifetime years, the values are as shown on page 27 of Exhibit 519.

Our CANDU units have a lifetime capacity factor of 73.3 per cent. All CANDUs, ours and CANDUs from other utilities, have a lifetime value of 74.2 per cent.

For the other reactor types, pressurized water reactors have a lifetime value of 65.5 per cent; boiling water reactors, lifetime value is 61.9 per cent; and gas-cooled reactors come in at 45.8 per cent.

Q. And who are the notably good lifetime performers?

A. There are 36 units rated at over 500 megawatts in the world which have individual lifetime capacity factors of 80 per cent or more. Ten of these

36 units are Canadian CANDU units: Point Lepreau in New Brunswick, all eight of our "B" units, and one of our "A" units.

Point Lepreau, which is a CANDU 6, has averaged 91.1 per cent over its nine-year life and is second in the world on a lifetime basis.

Based on lifetime performance we have five units in the top ten in the world: Pickering Units 5, 6 and 7 and Bruce Units 5 and 6.

In addition to the 10 Canadian units, other countries well represented in the top 36 are Germany with six units, the United States with five units, Belgium with four units, Japan with three units, and Spain with three units.

Q. All right. Finally, Mr. Daly, how would you describe the nuclear performance to date, and are you satisfied that Hydro's existing nuclear stations will continue to perform reliably over the 25-year planning period?

A. We have over 200 reactor years of learning from experience in operating, monitoring and maintaining all major nuclear plant equipment.

The experience we have gained in operation, maintenance and retubing of our older "A" stations is being transferred to our "B" stations and

is resulting in improved performance on those units.

There are some significant problems to be overcome, particularly at Darlington, but I believe we are putting in place the programs to deal effectively with these problems.

Finally, there are a number of broad-based programs in place that will serve to improve equipment performance, resource levels and management controls. These have all been developed to seek continued improvements in nuclear plant quality.

So yes, I am satisfied that our existing nuclear stations can operate reliably over the 25-year planning period.

Q. Thank you, Mr. Daly.

Turning now to you, Dr. Whillans. Dr. Whillans, as I noted earlier, will be providing evidence on the health effects of nuclear generation.

To begin with, what is ionizing radiation?

DR. WHILLANS: A. Well, simply put ionizing radiation refers to certain high-energy subatomic particles and waves that can travel through space from a source and have sufficient energy that when they interact with materials such as living tissue they cause ionizations, leading to physical changes in

the structure of the materials that can affect function.

Q. And what are the main sources of ionizing radiation in our environment?

A. Ionizing radiation exposures arise mainly from two kinds of sources.

THE CHAIRMAN: I wonder if you would just mind trying to speak just a little slower? It's a little hard to follow you. Have you got your microphone on?

DR. WHILLANS: Yes, I do.

THE CHAIRMAN: Perhaps if you could get a little bit closer to it it might help.

DR. WHILLANS: Sorry.

Ionizing radiation exposures arise mainly from two kinds of sources: first, from the admission of subatomic particles, such as alpha particles and beta particles, or of high-energy electromagnetic waves known as gamma rays from the nucleus of an unstable or radioactive atom as it decays; and secondly, by the deliberate bombardment of materials with these high-energy particles to generate secondary x-rays, such as those that are used in diagnostic radiology.

Every individual on earth is exposed to some kind of natural ionizing radiation.

The overhead, page 28 of our Exhibit 519 which is derived from our Exhibit 507, shows that on average this level of exposure is about three millisieverts per year.

I should note that in my evidence I will follow the new international system of radiological units whereby the unit of radioactivity is the becquerel, corresponding to one disintegration per second, and the unit of effective dose is the sievert. A millisievert shown on this figure is one one-thousandths of a sievert.

Now, the older system of curies and rems is still quite common, especially in the United States and in Ontario Hydro's working environments.

The most important conversion factor to remember is shown here. One sievert is 100 rem.

Now, returning to the question of average background doses, the three millisieverts is composed of about .3 millisieverts from cosmic radiation arriving from outer space, about .3 millisieverts from the decay of naturally occurring radioactive materials in the earth, and about .4 millisieverts arising from the decay of other naturally occurring radioactive materials, mainly potassium 40, within our bodies.

An additional and quite variable 2

millisieverts arises from breathing, again naturally occurring, radon gas evolving from beneath the earth's surface. The very substantial contributions of radon exposure to average background doses have been recognized only in the past several years.

Now, in addition, individuals may receive exposure contributions from a variety of artificial sources; for example, during medical diagnosis and treatment, or from consumer products. The values shown here are clearly just averages. Many individuals will receive no exposures for several years, but, on the other hand, some exposures will involve quite substantial doses.

Finally, at the bottom of the table there are shown some other much smaller artificial sources of exposure, such as those due to nuclear power generation or the residual activity remaining from atmospheric weapons testing mainly in the 1950s and 60s. Much of my later evidence will be directed toward demonstrating that the radiation dose impacts of nuclear power generation are indeed small.

MS. HARVIE: Q. Moving on, perhaps you would explain briefly what are the health effects that can occur as a result of exposure to ionizing radiation.

DR. WHILLANS: A. Well, a convenient basis for describing these effects is that provided by the ICRP.

Q. Excuse me. What is the ICRP?

A. The ICRP stands for the International Commission on Radiological Protection, and it is an international, non-governmental body that arose from 1928 as a branch of the International Congress of Radiology, a medical organization.

It assumed an independent existence after World War II with the growth of the peaceful uses of atomic energy. It is composed of experts in a variety of areas of radiation protection from around the world and publishes advice both in the form of general recommendations and does very specialized reports. Its advice has been adopted into the National Radiation Protection Legislation of most industrialized countries.

Q. Then, returning to the question about health effects that I posed a moment ago, would you continue, please?

A. Well, the ICRP divides these effects into two broad categories which it terms stochastic and deterministic.

And I refer you to the figure on page 29.

Stochastic means random, and stochastic effects are those effects associated with rare but significant changes for the exposed individual.

When ionizing radiation interacts with tissues the deposition of energy is not uniform but occurs in discreet isolated events so that even at very low average doses over the whole tissue sufficient energy can be deposited in critical sites to cause changes in function.

At the top of this figure are two representations of sensitive biological structures in a living organism.

Since the structure with respect to stochastic effects are probably single cells, in the diagram the cell to the left is shown to contain a single target, which, if altered, could change the cell's functions significantly. Such a target could be a part of the DNA, the program structure of the cell. The target may be small so that it is unlikely to be hit.

Also, since mankind has always been exposed to ionizing radiation biological processes have evolved which can repair most important target damage, but when unrepaired damage remains in a critical target molecule, such as the DNA, gross effects on the whole

organism could result.

Please refer to the lower half of the figure.

Stochastic effects are those effects for which the probability of effect occurring, but not its severity, depends on dose. Most important stochastic effects are the induction of cancers and genetic changes that will be expressed in succeeding generations.

For example, the consequences of a cancer are the same whether it has been induced by a large or small dose of radiation. For these stochastic effects it is assumed that even small doses of radiation can affect changes.

Now, this assumption has been made because it is presently impossible to distinguish between, for example, cancers that may have resulted from radiation exposure and those that arise from other causes. As a result, it is presently impossible to demonstrate a dose response relationship at the low doses that result from environmental exposure, and, in particular, to demonstrate the presence of a threshold or a level below which no effects would occur.

Also, because of this discreet deposition of energy that I referred to before for mechanistic

reasons it seems not impossible that even small exposures may effect changes. In Ontario Hydro risk calculations, therefore, we accept the recommendations of the ICRP that a proportionate risk of stochastic effects remains to the lowest levels of dose as shown on this figure. This means that when large populations are exposed even to very low average doses some very few individuals may develop cancers or genetic damage. This is known as the "linear hypothesis".

Q. What about the other kind of effects, the deterministic effects?

A. Well, deterministic refers to an outcome that results not from single random hits but from a sequence of continuous, incremental steps.

With respect to human exposures, these are effects that occur only at very high doses, in the order of sieverts as I have indicated here, a thousand times or more greater than what individuals would receive from background, as I showed on the previous overview, or from normal occupational exposures. They are of concern to us now mainly in accident situations.

As I have indicated on the right-hand side of this figure, for deterministic effects the sensitive biological structure may be a whole organ containing many targets, individual cells.

Large doses of radiation produce sufficient damage in a cell to cause its death. Death of one or a small number of cells in a tissue is usually of no consequence. The tissue just repopulates and recovers.

However, when sufficient fractions of the cells are damaged within a short space of time changes in the function of the organ will be detectable. The level of change, and therefore dose required to cause detectable change, constitutes a threshold, and this threshold depends both on the organ and the tissue or on the level of injury regarded as important.

The very bottom right-hand portion of the figure shows continuously increasing severity of damage in the tissue with dose and also a threshold beyond which functional changes will be detectable, which in this case is about one 1 sievert.

As dose increases beyond the threshold the severity of the effect and therefore the probability of total organ failure increases.

An important example of a deterministic effect is radiation skin burns. Small or even quite large exposures to the skin up to several sieverts result in no observable changes in the skin.

Beyond about six to eight sieverts,

however, a mild arethema or reddening would occur, and as the dose increases the severity of damage increases until there is complete tissue breakdown.

Exhibit 507 provides on its final page a table of thresholds for the most important radiation-sensitive tissues in the body.

The most important radiosensitive tissue is the bone marrow with a threshold of about one sievert. Its failure produces a life-threatening situation and largely determines the so-called LD50, the dose that is lethal to about 50 per cent of exposed individuals.

In humans this dose lies in the range of 2-1/2 to 5 sieverts, depending on the medical treatment available, somewhat greater than the value I have illustrated on this figure.

So to summarize, there are two main classes of health effects, stochastic and deterministic.

Deterministic effects have thresholds much greater than any doses that would occur under any normal public or occupational exposure conditions. They are mainly of concern in medical treatment where large doses must be given or for accidents.

Stochastic effects may, in principle,

occur down to the lowest levels of exposure but with increasingly small frequency.

Q. Dr. Whillans, how do we know about these risks of radiation exposure?

A. Estimates of the risks of radiation exposure to humans are derived from three main courses: the health experience of the survivors of the atomic bombings of the Japanese cities of Hiroshima and Nagasaki in 1945; the results of diagnostic and therapeutic medical exposures; and the experience of populations exposed occupationally or environmentally. [12:25 p.m.]

The first of these has been by far the most influential.

The population exposed in Japan contained a full age distribution of both sexes, was exposed to a wide range of doses from very low to a lethal level, and has been followed now for more than 45 years.

There are of course deficiencies in relying on this data alone; for example, the population was exposed acutely, largely within fractions a second. And we now believe that the prolonged exposure, such as would occur in an occupational or public environment, may be less damaging.

So the other populations do contribute to

our knowledge of these risks, but in general, these other studies have much smaller collective doses and often have other methodological problems that limit their usefulness.

The health experience of these populations is reviewed periodically by a number of independent bodies, most notably by UNSCEAR, the United Nations Scientific Committee on the Effects of Atomic Radiation, the BIER committees, committees reporting to the U.S. National Academy of Sciences which are assembled periodically to review to the biological effects of ionizing radiation, and by the ICRP.

The most recent update of the Japanese survivor data to 1985 has been reviewed in UNSCEAR in its 1988 publication, in BIER 5, 1990, and also by the ICRP for its recent recommendations in publication 60, published in 1991. All three come to similar conclusions regarding the risks of radiation exposure.

I should also comment at this point on the information that has come from Chernobyl accident. I am sure we are all aware of the disastrous accident that occurred in 1986 at the Soviet reactor complex at Chernobyl. At least 31 workers died immediately as a result of very high radiation exposures and burns received fighting the resulting fire, and fairly large

amounts of radioactivity were blown out of the reactor and carried over local and even distant populations.

At this time, I would simply point out that studies to date of populations exposed as a result of the Chernobyl accident have not changed our estimates of risk.

First, the period since the accident is still too short to expect to have seen any significant effects on cancer rates, and preliminary epidemiological studies do confirm this.

An important problem however is the very primitive state of the Soviet health care system in the region, and especially with respect to health records, so that in general only unscientific, anecdotal reports coming out of a politically charged environment have been available.

Several international studies are being organized to follow these populations, and I will return to this later.

Q. Dr. Whillans, how do these new estimates of risk compare with previous estimates?

A. The last general review of risks published by the ICRP were in 1977, and it reviewed data from 1945 to about 1974. Since that time there have been two major developments that affect the risk

estimates.

First, the doses to the survivors in Hiroshima and Nagasaki have been re-evaluated and found to be lower than had been previously estimated, on average only about 50 to 70 per cent of the values estimated in 1965.

This lower estimate of the exposures that were responsible for the observed effects leads immediately to an increase by a factor of one-and-a-half to two in the derived risk per unit dose.

The second development is that after 11 additional years of follow up, a higher number of particularly solid cancers has been seen than was predicted in 1977.

It should be remembered that even in 1985, nearly 60 per cent of that study population was still alive, and so projections must be made in order to estimate lifetime risks.

These projections are, of course, subject to uncertainty, and by 1985 more cases had been seen than were earlier predicted. However, recent data indicate a downturn in the number of new cases appearing, and there is now greater confidence in the present projections.

From this additional follow up data it's now more clear that the appropriate method of projection is based on a relative risk rather than an absolute risk model. This means that the increased risk as a result of a given dose of radiation is a multiplier of normal cancer risk rather than being a simple addition to that normal risk. The results of this change and projection methodology lead to a further increase in the risk estimates.

The figure on page 30 provides a summary of the changes in the risk estimates for stochastic effects recommended by the ICRP in 1991 as compared to 1977, averaged over age and sex and provided both for workers and for members of the public.

The risk of inducing a fatal cancer is now estimated to be 4 to 5 per cent per sievert, depending on the population being considered, or in units more relevant to normal nuclear power generation exposures about 0.005 per cent per millisievert.

Overall, the new information leads to an estimate of the risk of cancer about three or four times that accepted in 1977 when only a single estimate was provided.

Q. What about the other stochastic effect --

THE CHAIRMAN: Perhaps if I could interrupt. We should perhaps stop now and pick this up at 2:30.

THE REGISTRAR: Please come to order. This hearing will adjourn until 2:30.

---Luncheon recess at 12:30 p.m.

---On resuming at 2:35 p.m.

THE REGISTRAR: Please come to order. This hearing is again in session. Be seated, please.

MS. HARVIE: It was just brought to my attention, the fuel bundle has gone missing. I hope no one has decided to take it home as a souvenir.

MR. NUNN: It still here.

MS. HARVIE: It's still in safekeeping.

MR. NUNN: Yes.

MS. HARVIE: Q. Dr. Whillans, what about the other stochastic effect, the genetic risk?

DR. WHILLANS: A. Well, 30 years ago radiation dose limits were set on the basis of concern about the accumulating damage to the world's genetic pool as a result of radiation exposures.

At that time it was recognized that radiation exposure also caused cancers, particularly leukaemias but then only 15 years after the bombings in Japan, the extent of that risk could be only

incompletely estimated.

The more serious concern about genetic damage was based on laboratory studies going back to the 1930s with fruit flies and some limited studies in mice.

Now after 45 years, and when other exposed human populations have been studied, the view is somewhat different. Genetic changes attributable to radiation exposure have never been observed in man, for example, there is though detectable increase in genetic defects in the more than 30,000 children born to parents who received average gonad doses of more than 400 millisieverts in the bombings of Hiroshima and Nagasaki. If the risks were as great as those estimated from the laboratory studies, they should have been seen.

However, those extensive laboratory studies do demonstrate mechanisms which most experts agree must operate to some extent in man. Therefore, the estimates of genetic risk derived from animal experimentation, and which are believed to be conservative when applied to man, are included in the ICRP's total estimates of risk as shown in the figure.

As can be seen, these estimates have not changed significantly since 1977 when a range of risk

was recommended. This situation reflects the lack of significant new knowledge over the period and the continued reliance on laboratory results.

Genetic risk, however, is now relatively less important in relation to cancers, only about 10 to 20 per cent of the total risk.

Q. Dr. Whillans, aside from the risks of inducing fatal cancers and genetic risks, are there any other health effects of special concern that are associated radiation exposure?

A. Yes. Of additional concern are the adverse effects that might arise as result of in utero exposures; that is when the exposure occurs prenatally to an embryo or fetus.

Overall, the evidence points to three distinct kinds of risk: The early loss of a pregnancy, developmental damage to the brain, and induced cancers.

First, it's known from animal studies that very high doses to an early embryo within the first days after conception results in the loss of pregnancy. However, the animals that survive these early exposures appear to be unaffected since it seems that any significant damage at this stage causes the pregnancy to terminate.

There are no corresponding observations

in the human studies since the loss would usually occur within the first one or two or weeks before the woman might be aware of her pregnancy.

The doses at which pregnancy in animals are lost much higher than those than would be normally received, even in occupational environments, the order of some tens of millisieverts. This form of risk is generally of less concern than those that I will discuss now.

Second, animals are radiated in the laboratory to high doses later in gestation have been observed to develop malformations, especially defects of the skeleton. Although skeletal defects have never been associated with human in utero exposures, a related phenomenon of small brain size and severe mental retardation has been demonstrated in the children of Japanese mothers radiated to high doses and mainly in the period 8 to 15 weeks after conception. The literature provides great detail about this risk which is summarized in our Exhibit 507.

The probability of inducing severe mental retardation at high doses is quite large, about 40 per cent per sievert, but only in the period 8 to 15 weeks after conception with a lesser risk extending perhaps to 25 weeks. This is the period of maximum development

of the cerebral cortex. No detectable risk occurs at earlier times before eight weeks and especially in the period when the woman may not yet be aware of her pregnancy.

Third and finally is the question of induced cancers.

More than 30 years ago Dr. Alice Stewart reported an increased risk of childhood cancers, mainly leukaemias, among children whose mothers received diagnostic X-rays during pregnancy.

Subsequent studies of other medically radiated populations have tended to confirm this result, and the risk experienced by the approximately 1,600 children exposed to in utero at Hiroshima and Nagasaki are only marginally smaller.

The reports by UNSCEAR and BIER conclude that the risk of inducing cancer by in utero radiation exposure is probably several fold greater than that for and an adult, but similar to that for radiation during childhood.

Q. How have these risk estimates been used to determine individual radiation exposure limits for both workers and for the public?

A. First, I would like to point out that individual dose limits are only one part of the system

of radiation protection recommended by the ICRP and adopted in many countries.

The ICRP system is based on three concepts. The first is justification, which means that because we accept that any radiation exposure may carry some risk, any activity which results in such exposures must be justified.

If the net benefit of the activity, for example, producing isotopes for medial use, doesn't exceed the cost and including risk to workers to workers in the public, it's not justified.

Second, the method of carrying out an activity should be optimized with respect to radiation risk.

Here the guiding principle is to keep all exposures as low as reasonably achievable, commonly known as ALARA, economic and social factors being taken into account. This means that additional barriers to exposure or other kinds of protective controls must be added until other social or economic factors make it unreasonable to do so.

In exercising its regulatory responsibility, the AECB focuses a large part of its efforts on ensuring that this ALARA process is followed.

Finally are the controls about which you asked, individual dose limits. The purpose of these limits is to ensure that in carrying out an activity that would result in radiation exposure, the distribution of doses and therefore radiation risk among individuals does not result in an unreasonable burden on any single worker or group of workers or on any group of members of the public.

Q. What are these individual worker dose limits and how are they arrived at?

A. The limits recommended by the ICRP and adopted international regulations including those of Canada are based on the principles of avoiding all deterministic impacts entirely and of keeping the risk of stochastic effects, cancer, inheritable damage within acceptable bounds relative to other accepted risks.

In its most recent recommendations published as publication 60 in 1991, the ICRP provides 200 pages of detail on this process.

The limits are different for radiation workers as compared to the general public, not just because workers derive direct economic benefits from their work but also because the public includes the most sensitive members of the population, such as

children, and public exposures will continue for longer periods of time.

The figure on page 31 summarizes the main features of the current Canadian annual dose limits and compares them to those recommended by the ICRP both in 1977 and in 1991. The limits are shown both for the whole body and also for other individual organs.

Recall that the thresholds for deterministic effects, even when the exposure is received acutely, are at least 1,000 millisieverts, that is one sievert, and that when exposures are prolonged over weeks and months, the thresholds are generally much greater.

First let us discuss the dose limits for radiation workers, at the top.

The present Canadian limits are based on ICRP recommendations going back over 25 years. In 1977 in its publication 26, the ICRP produced a much more comprehensive analysis of radiation risk, and related the risk accepted under those recommendations to the risks of acute accident fatalities in what are generally regarded as safe industries. They also reviewed the evidence on acute; that is to say deterministic effects, and concluded that a single somewhat higher limit of 500 millisieverts per year for

single organs with exception of the lens of the eye, it retains a lower limit of 150 millisieverts, but for the other organs though a limit of 500 millisieverts was more than adequate to prevent those effects.

Now, although the basis for the 1977 recommendations was more solid, in fact the recommended limits were in some cases less restrictive than the earlier values. For this reason there has been no urgency to adopt these recommendations in the U.S. or in Canada, and in fact both countries are just now revising their legislations.

In 1991, however, the ICRP responded to the new estimates of stochastic risk that we discussed earlier, by lowering its recommendation for the principal occupational limit to the whole body from 50 to 20 millisieverts per year with some provision for averaging doses over longer periods.

In Canada the AECB has already responded to ICRP publication 60 by proposing in its consultative document C122 a new set of limits and issuing them for public comment.

The legal process necessary to amend this legislation, however, is lengthy because there are many issued to be weighed. It is possible that the legal limits in Canada will not change to meet these latest

ICRP recommendations for some years.

THE CHAIRMAN: What do they propose? You said they have proposed new limits.

DR. WHILLANS: I'm sorry, I am having trouble hearing.

THE CHAIRMAN: I thought you said they proposed new limits. I wondered if you could quantify that.

DR. WHILLANS: The limits proposed in the AECB document, C122, are basically those recommended by the ICRP in its 1991 publication. So a principal limit of 20 millisieverts.

THE CHAIRMAN: Thank you.

DR. WHILLANS: Although this legal process is under way, Ontario Hydro has already reacted to the new knowledge by introducing its own, more restrictive dose control guidelines. An internal policy signed in 1991 sets targets for maintaining individual exposure, worker exposures below 20 millisieverts in any single year with a long-term average over five years of 10 millisieverts.

MS. HARVIE: Q. Those are the occupational dose limits. What about the public dose limits?

DR. WHILLANS: A. This figure also

includes a general summary of public dose LMSTM.

In general, public dose limits have been recommended to be a factor of 10 lower than worker limits, the reasons I discussed earlier.

The present legal limit in Canada for whole body exposure of a member of the public, apart from background and medical exposures is 5 millisieverts per year, as is shown in the bottom part of the overhead.

Partly on the basis of this new risk information, the ICRP has recommended reducing this value to 1 millisievert, that is to about one-third of the dose due to natural background. As I have said, changes in Canadian legislation are in progress.

I would emphasize that the limit applies to the maximally exposed member of the population, the so-called critical group, with respect to any activity, and the exposures to most members of the public as a result of these activities would be lower.

Moreover, Ontario Hydro typically controls its emissions such that exposures even to the critical group are about 1 per cent of the dose limit, or about .05 millisieverts. I will discuss this evidence in more detail later.

Q. Aside from the occupational and

public dose limits, are there any other dose limits that the Board should be made aware of?

A. Not really dose limits, but a set of reference levels for deciding on the actions to be taken should an accident occur with potential for exposures to the public.

In Ontario an emergency response plan, a copy of which was provided in our expense to Interrogatory 9.17.34, is in place to respond to any nuclear emergency, despite the very low probability that such an emergency will occur. Mr. King will be talking in more detail about this plan in the context of nuclear safety later.

This plan, which is similar to plans in place in the U.S. and the U.K., is structured about a set of protective action levels or PALS, as shown on the figure on page 32. Two sets of PALS, an upper PAL and a lower PAL are shown. For each of three different intervention measures, sheltering, KI or potassium iodide pill administration to block uptakes to the thyroid of radioactive iodines, and full scale evacuation.

The doses shown on this overhead are generally whole body doses except for the middle row where the dose to the thyroid is of primary concern.

Each intervention is considered by the provincial response organization when the projected dose that may be received by any sector of the population exceeds the lower PAL, and action is mandatory above the upper PAL. Between these levels the province would apply judgment taking into account other factors.

As can be seen, these action levels range from an additional dose of one millisievert, again only about one-third of natural background, for simple actions which carry little risk, to a whole body dose of 100 millisieverts, twice the annual occupational limit, but still well below the level at which any acute affects that would result, that is that some thousands of millisieverts.

Q. Dr. Whillans, you have told the Board about the health effects that can result from exposure to ionizing radiation, we have heard about how a knowledge of the risks of these health effects has been used to derive the legal limits for exposures to workers and to the public. Can you please describe how Ontario Hydro's performance in controlling doses and any measures of the impacts of these exposures on health.

[2:50 p.m.]

A. Ontario Hydro has operated a nuclear program since the 1960s with commercial service at Pickering beginning in 1971. Throughout this period Ontario Hydro has always operated to legal dose limits that would be applicable today and with very few excedances, which I can describe later.

Future occupational dose limits in Canada following the new recommendations of the ICRP, and present Ontario Hydro guidelines, however, are more restrictive, and so in order to provide the Board with a basis for predicting future performance I refer mainly to our current record and that of the recent past.

The figure on page 33 shows the total annual collective dose received by all Ontario Hydro workers for each year in the 10-year period 1981 to 1990. These data are derived from the 1990 Ontario Hydro Annual Dose Summary sent out in response to Interrogatory 9.17.37.

Collective dose refers to the total dose received by the whole population of exposed workers, summed over individuals, and expressed in units of person-sievert. Exposed workers here refers to any workers who were monitored for and received a measurable dose of radiation.

This annual collective dose, now about 15 person-sievert per year, has not increased over the 10-year period despite the fact that Pickering "B" came on line in 1983 and Bruce "B" in 1984, nearly doubling the existing generating capacity and also that there have been major rehabilitation efforts at Pickering "A".

In fact, were these data normalized to give a value per unit generation one would see a reduction by about a factor of two over this 10-year period as was shown in our response to Interrogatory 9.17.37.

DR. CONNELL: Could I just ask if these data include natural and medical as well as in plant?

DR. WHILLANS: No.

DR. CONNELL: Just in plant?

DR. WHILLANS: These are just occupational exposures.

DR. CONNELL: Thank you.

MS. HARVIE: Q. How does this performance compare with other nuclear generation facilities?

DR. WHILLANS: A. The figure on page 34 compares Ontario Hydro's performance over the five-year period 1985 to '89 with that in other countries, as was

described in our response to Interrogatory 9.7.206.

Ontario Hydro's total occupational exposure per unit generation is similar to those of the U.K. and Sweden, lower than those of Japan, France and West Germany, and more than a factor of three lower than that of utilities in the U.S.

Q. A minute ago you mentioned a value of about 15 person-sieverts. What does this mean in terms of risk?

A. Using the most recent values recommended by the ICRP, 4 per cent or four by 10-to-the-minus-2 fatal cancers per sievert, as was shown on a previous overview, and on the assumption that a proportionate risk extends to these low levels exposure this collective dose translates to 0.6 fatal cancers induced in each year of operation in about 5,000 workers. These cancers would be expected to appear over the workers' remaining lifetimes.

In comparison, the number of fatal cancers that would be expected to occur as a result of other causes over the lifetimes of a population of this size is about 1,000.

Q. And what about doses to individual workers?

A. The figure on page 35 shows the

average annual dose to Ontario Hydro workers over the same 10-year period 1981 to 1990 as derived from the same source.

As can be seen, there is a general downward trend for the 1990 value of about 2 millisieverts per year, and this represents the results of deliberate dose reduction efforts by Ontario Hydro and its workers. This value should be compared to the likely future worker dose limit of 20 millisieverts and the present value of 50 millisieverts.

Q. These are average values. Do some workers receive much higher doses?

A. While it's true that there is a distribution of worker doses and some workers do receive higher exposures, even these higher doses are usually well within the legal limits and apply to a fairly small percentage of workers.

The figure on page 36 shows the distribution of individual doses in 1990 by dose interval, again derived from the same report.

On the vertical axis is shown the percentage of the radiation workers exposed to a given dose interval and on the horizontal axis is shown the average dose in that interval.

As can be seen, the great majority of

individual doses lie below 10 millisieverts and over the past five years only about 60 workers per year of the 5,000 exceeded 20 millisieverts. Of course, these workers that are the focus of our current dose reduction program.

Q. In future years it is likely that there will be an increasing need for reactor rehabilitation, such as the retubing work at Pickering "A", as we heard from Mr. Daly.

Do these jobs result in higher radiation exposures and routine operations?

A. No, not on average. It is true that as we first entered into rehabilitation work we encountered new radiological situations that resulted in higher exposure levels. These accounted for the peaks in both the individual and collective dose figures that I showed previously around 1983. These doses have, in general, been reduced with experience.

The average dose to our attached staff, which are mainly construction contractors for the rehabilitation work, was 2.4 millisieverts in 1990 compared to the overall radiation worker average of 2 millisieverts, and, in fact, over the last five-year period the average dose to attached staff was 3.1 millisieverts compared to the total worker average of

3.5.

Q. Again, the numbers that you have been discussing are mainly average values over time, and there is, of course, concern about unplanned events such as when high doses can be received.

Can you give us those figures, please?

A. Yes. From time to time events do occur where workers receive higher doses than had been planned and certainly higher than can be considered acceptable. For example, in 1989 two workers at Pickering exceeded the annual legal dose limits as a result of improper shielding. However, these events are relatively rare.

Since 1963 there have been only 18 occasions on which an Ontario Hydro worker exceeded the annual whole body dose limit of 50 millisieverts, less than one event per year.

In comparison, in 1990 alone, for example, 16 radiation workers in Canada exceeded this limit, proportionately a much greater number, and in 1990 this included two dental hygienists.

The maximum whole body dose received by an Ontario Hydro worker has been 129 millisieverts. That is about 2-1/2 times the dose limit. This is a completely unacceptable exposure, but nevertheless, one

well below the level at which any immediate deterministic effects would be seen.

Q. Dr. Whillans, can you describe how Ontario Hydro is planning to reduce worker doses in the future?

A. Well, there are many approaches Ontario Hydro is following to reduce occupational radiation exposure, most of them under a large umbrella program called QIP, Q-I-P, for Quality Improvement, referred to earlier by Mr. Daly.

This program was described in our response to Interrogatory 9.2.122. The QIP program is studying everything from work planning methods to the use of protective equipment.

Another more specific approach is the construction of the tritium removal facility, or TRF, at Darlington. Tritium is a radioactive by-product of the CANDU generation process. Tritiated water uptakes from the air in our stations has accounted for about half of the total occupational dose over the past several years.

The tritium removal facility processes the heavy water which contains the tritium from each reactor unit and extracts most of the tritium before returning the heavy water to the system; thus, the

source of significant occupational exposure and also tritium emissions to the environment is much reduced.

Q. Can you turn now to the results of the studies to detect adverse effects in radiation workers. What studies has Hydro carried out in this regard?

A. Ontario Hydro for many years has contracted with Dr. Terry Anderson at the Department of Health Care and Epidemiology at the University of British Columbia to review mortality records of all Ontario Hydro workers and pensioners.

His reports, covering the period from 1970 to 1988, were provided in response to Interrogatory 9.22.32 and show that the mortality of our nuclear workers from all causes of death is only 60 per cent of that from an age-matched sample of the general population. This is the so-called "healthy worker effect" observed in many occupational studies and is attributable to the selection of healthy individuals for work. However, a similar value of 62 per cent applies to total cancers.

Now, over this whole period only 36 Ontario Hydro nuclear workers or pensioners have died of cancer so that a finer examination of these records, for example by cancer site or by received dose, is very

difficult.

It must be acknowledged also that most radiation exposures in Ontario Hydro are relatively recent, on average only about 10 years ago and less than the latency period for many solid cancers. However, the data do show that there has been no detectable increase in cancers or any other causes of death in our radiation workers.

Q. What about the experience at other Canadian facilities?

A. Well, for example, the Chalk River laboratories have been operating since the 1940s and have accumulated a larger number of exposures over a longer period of time.

THE CHAIRMAN: Would you please slow down? The reporter is having trouble taking what you --

DR. WHILLANS: Sorry.

THE CHAIRMAN: Just try and read a little slower. I know it's difficult, but just try and read a little slower.

DR. WHILLANS: I will repeat what I have just said.

The Chalk River laboratories have been operating since the 1940s and have accumulated a larger

number of exposures over a longer period of time.

The mortality of their workers has been reviewed by their own health statisticians and also by Dr. Jeffrey Howe of the University of Toronto. The report of a recent mortality study was provided in response to Interrogatory 9.22.30. Their findings are similar to those at Ontario Hydro. There is a healthy worker effect and no evidence to date of an increased risk of cancers.

Again, however, this is a relatively small population.

Q. Are there any other studies of occupationally exposed populations that can provide better evidence?

A. The problem is mainly one of size for these populations, which even in the early days received relatively small exposures compared for example to many of the Japanese survivors.

Studies have been reported of worker populations at the Hanford site in the U.S., at other American facilities, and from the U.K. Atomic Energy Authority establishments.

Overall, while some individual studies have reported significantly elevated and some significantly depressed risks for individual cancer

sites these results vary from study to study without consistency and could have occurred by chance.

The International Agency for Research on Cancer located in France is attempting to address this problem by pooling dose and health experience data from many of the countries with extensive worker exposures into a large, single analysis. This study is expected to be reported in about two years.

One very recent study that deserves mention is the report in January of this year from the National Radiological Protection Board of the U.K.. Their study, looking at workers on the U.K. National Dose Register, has found a significantly elevated risk of leukemia with increasing dose.

Now, this method of looking by dose category within the study population requires large numbers of records but where sufficient numbers are available is a much more sensitive method of analysis because it corrects for the healthy worker effect.

Because this is still a preliminary result and a larger analysis using more radiation worker records will be completed in about one year the errors on the present estimates are still large.

The central risk estimate derived from the present data is about a factor of two greater for

leukaemia than that recommended in ICRP Publication 60, but the ICRP estimate of is well within the band of error.

No other cancer sites were significantly elevated. Moreover, the results for leukaemia are somewhat at odds with those found in a comparable American study. The author's conclusion is that their data justified no change in the ICRP recommendations, but we will, of course, follow the results of their later studies and those of the international agency carefully.

Finally, though, the NRP -- the results do give us increased confidence that the ICRP risk estimates do not underestimate the true risk by a factor of 10 as has sometimes been claimed.

Q. Perhaps we can move now to public health performance. What are the potential sources of public harm as a result of nuclear power generation?

A. The main sources of potential harm are usually accepted to be the radioactive emissions from nuclear generating situations, although other emissions from associated Ontario Hydro facilities, such as hydrogen sulfide from the Bruce heavy water plant, are also of concern. These will be discussed by Mr. Johansen.

In addition, radioactive and other emissions from other parts of the fuel cycle, such as from mining and the fabrication of fuel and the long-term disposal of waste, were considered in some detail in our Exhibit 507.

Q. Can you tell us briefly how public exposures from Hydro's nuclear stations are controlled?

A. With respect to particularly radioactive emissions, the controls begin with the legal public dose limits that we discussed earlier. Based on ICRP recommendations, no member of the public should received more than currently 5 millisieverts per year as a result of industrial sources of exposure.

Now, in order to relate this limit to the control of emissions from a plant it is necessary to study on a site-specific basis the distribution of a population that might live or work in the vicinity of the site and to identify the various environmental pathways that could lead to exposure of this population from the emissions.

A so-called critical group is then identified which represents the subset of the general population that would receive the largest exposure by each of the various pathways and an emission limit known as the Derived Emission Limit, or DEL, is

established to keep these potential exposures below the limit.

Ontario Hydro's development of DELs is described in detail in our response to Interrogatory 9.22.6.

Throughout the process, which must be approved by the AECB, many conservative assumptions are made to ensure that the dose limits will not be exceeded.

Q. Now, what has been Ontario Hydro's performance with respect to meeting these limits?

A. Ontario Hydro has set an internal target of 1 per cent of the DEL as a limit for its releases, and this target is referenced in its station licenses. Over the years emissions have almost always met this target.

The figure on page 37 shows an example taken from our 1990 annual summary and assessment of environmental radiological data provided in response to Interrogatory 9.17.36. This is a very busy figure which shows annual releases over the past five years for each of the six major categories of emissions from Pickering "A", which is the oldest station, has generally the highest emissions.

The important point to note is that on an

annual basis these releases are almost always below, sometimes much below, the 1 per cent target shown by the dotted lines. Results for all of our other facilities are generally similar.

I would add that although the summary of 1991 emissions is not yet available, the weekly and monthly data show no change in this pattern.

Q. We see here though several different pathways, and besides, there are two adjacent stations, Pickering "A" and Pickering "B", whose emissions presumably would expose the same population.

Is there any more direct evidence about exposures actually received by the public?

A. Yes. Because the DELs are developed for individual radionuclide groups, and also because they are based on somewhat uncertain assumptions about environmental pathways, a direct monitoring program is also in place to detect the presence of these radioactive emissions in the air that's breathed, in the water that is drunk, and also in various other sources, such as milk and foodstuffs that people may eat.

The results of this monitoring program are also reported in the annual summary I referred to above, which is prepared by the Health and Safety

Division and submitted to the Atomic Energy Control Board.

Part of this assessment is the calculation of doses to critical groups based as much as possible on direct environmental measurements.

The figure on page 38 shows an example. It is a table from the 1990 report showing calculations for Pickering which include emissions from both stations and all pathways. Calculations shown here have been carried out, as you can see in the three vertical columns on the right, for several potential critical groups.

For example, infant "A" on the left drinks fresh local milk which may contain radioiodines, whereas infant "B" drinks powdered milk which was reconstituted from local tap water that may contain tritium.

From calculations such as this it can be seen that the highest critical group dose in this area was about 44 microsieverts; that is, .044 millisieverts in 1990, less than 1 per cent of the public dose limit of 5 millisieverts. Results at the other stations are similar.

I would emphasize again that this dose, which is less than 2 per cent of the natural background

exposure received by the same population, is for the critical group, and the exposure of most of the population would be even smaller.

Q. Although the exposures of individuals are small, a fraction of background, isn't the fact that large populations are exposed of concern?

A. Yes.

THE CHAIRMAN: I'm sorry, could you just go over that again? I wasn't sure I followed your analysis from this table. What were your conclusions from this table?

DR. WHILLANS: Sorry. What is shown here is an example of a critical group calculation, in this case for Pickering for the year 1990, and what I expected to show is that there is a summing of doses to potential critical groups, the most exposed groups, from all the major exposure pathways.

THE CHAIRMAN: Right.

DR. WHILLANS: Various ingestion pathways, inhalation for various nuclides, skin absorption and external radiation by normal gases.

When these are all summed, for example for infant "A", the total annual dose is 44 microsieverts.

[3:10 p.m.]

THE CHAIRMAN: Where do I see that?

DR. WHILLANS: This is at the bottom of the second column.

THE CHAIRMAN: I see it.

DR. WHILLANS: 44.5, and the units are microsieverts, which are a 1,000th of a millisievert. And I converted that to .044 millisieverts, and said that this was less than 1 per cent of the public dose limit which is five millisieverts.

And the other critical group calculations are lower, so they are not critical in an absolute sense. A number of potential critical groups are evaluated and then a decision can be made about what the highest likely exposure would be.

THE CHAIRMAN: The limits are the same for infants as for adults, are they?

DR. WHILLANS: The limits are the same, yes, the whole body limits are the same.

MS. HARVIE: Q. Perhaps I will repeat my question.

Although the exposures of individuals are small, a fraction of background, isn't the fact that large populations are exposed of concern?

DR. WHILLANS: A. Yes. The measure of significance from a societal point of view should be

the total collective exposure integrated over the whole exposed population in units again of person sievert.

Such a number is also calculated annually from the estimates of exposure for various locations around the stations and the population distributions. These collective dose estimates are included in the annual summary to which I have referred.

The figure on page 39 summarizes these calculations for the most recent year 1990. Again, results for other years have been similar.

As can be seen at the bottom right-hand box, the total population collective dose as a result of all Ontario Hydro's generating facilities is about 1.75 person-sievert. The major contribution arises from the most densely populated area around Pickering.

Q. And what risk does this exposure correspond to?

A. Again, using most recent ICRP risk figures, a collective dose of 1.7 person-sievert per year corresponds to about 0.1 committed fatal cancers per year in the entire exposed local population of about 3 million sometime during their remaining lifetimes, and again assuming that a proportionate risk of cancer extends to these very low levels of exposure.

Q. Now, you have been talking about

Ontario Hydro's facilities only. What about exposures resulting from other aspects of the fuel cycle?

A. Exhibit 507 reviews these risks for all aspects of the fuel cycle. In general it's estimated that as a result of mining and milling or short-term waste storage, the most exposed individuals would receive a similar few per cent of natural background. However, since these facilities tend to be in isolated locations, the contributions to the population collective dose are small.

Final information for any future long-term storage facility that may be accepted are of course not yet available, but are also likely to be very small. In fact, adequate control to address these concerns is the reason for the long and careful development of acceptable methods.

Q. Dr. Whillans, would you turn now to discuss studies of public health in the vicinity of the nuclear generating stations. Have there been any such studies around Ontario Hydro's facilities?

A. Yes, and I would like to discuss with you the two principal studies. First, that of childhood leukaemia around Ontario's nuclear facilities, and second birth defects in the area around Pickering.

Q. What were the results of the first study concerned childhood leukaemia?

A. Well, because of concerns arising out of reports from the U.K., Atomic Energy Control Board sponsored a study of childhood leukaemia which was carried by the Ontario Cancer Treatment and Research Foundation, which focused around five Ontario nuclear facilities, the Ontario Hydro stations at Pickering and at Bruce, the region around the Chalk River laboratories, which also includes the former Ontario Hydro NPD station and two mining and fuel processing sites at Elliott Lake and Port Hope.

The study was carried out in two phases with reports published in 1989 and 1991. These reports were provided in response to Interrogatories 9.29.7, and 9.9.26 respectively.

The authors of these studies found no significant difference in the incidence of leukaemia in children at any of the sites since they began operation compared to Ontario as a whole.

Some of the relative risks were higher, for example, for the periods since 1971 in the area around Pickering the relative risk was 1.34 or 34 per cent higher than would be expected.

The risk was also somewhat greater than

one for the period before the plant began to operate.

Some of the risks were lower. For example, for the period since 1950 the risk around Chalk River was between .3 and .9.

All of these results, however, were based on relatively few cases and none achieved statistical significance, and in the opinion of the authors could have occurred by chance.

Q. What about the study of birth defects around Pickering?

A. This study is also discussed in Appendix 2 of Exhibit 507.

In 1988 a private citizen, Mr. David McArthur, claimed to have found a possible connection between releases of tritium from Pickering station and birth defects and infant mortality in the surrounding area. This analysis was questioned by Ontario Hydro and by the Ontario Ministry of Health and so a study was commissioned by the Atomic Energy Control Board and carried out by Health and Welfare Canada. This study reviewed all of the births defects data for the area from 1971 to 1988.

The Health and Welfare report published in 1991 by the AECB examined a variety of end points and concluded that the rates of infant deaths and

stillbirths in the area were not higher than in other Ontario communities and did not correlate with tritium releases.

The report also looked at 22 specific categories of birth defects and found only one, Downs Syndrome, that was statistically elevated. However, because there was no evidence of an association with tritium releases, the authors concluded that this result could not be attributed to tritium exposure and may have occurred by chance.

Q. Are there any other studies under way in Canada?

A. To my knowledge, one.

Although no statistically increased risk of childhood leukaemia was found around Ontario nuclear facilities, the cases identified in that research are being used as the basis of a case control study to determine whether these cases were more likely to have occurred to fathers who were radiation workers. This study also being conducted by the Ontario Cancer Treatment and Research Foundation and sponsored by the AECB is due to be released in 1992.

Q. How do the results of these studies fit in with other research that's being carried out in other countries.

A. Internationally, of course, these same issues are of concern and numerous studies have been directed at them.

Of special interest are the reports of increased risks of cancers, especially childhood leukaemia around the British facility Sellafield, formally know as Windscale which has operated a power generation, reprocessing and research establishment since the early 1950s.

In January of 1983 a television documentary reported an apparent large excess of childhood leukaemias in the nearby town of Seascale over the period 1950 to '83. Five cases were observed versus only 0.5 expected.

This excess was confirmed by a government inquiry but analysis by the National Radiological Protection board of the U.K. made it seem unlikely that environmental emissions could be responsible.

A subsequent case control study over the surrounding region by Dr. Martin Gardner of the environmental epidemiological unit of the University Southampton reported in 1989 a strong and statistically significant association with father's radiation work, but also an association with other kinds of non-radiation work. This result was not repeated at

other sites such as Dounray which also appear to show statistical increases in the population. Follow up studies are still under way to clarify these results.

The Sellafield studies accelerated efforts to identify similar risks around other nuclear facilities, particularly in the U.S., the U.K., and in France.

In the U.S., Jablon of the National Cancer Institute reported no excesses around 60 nuclear facilities, and a similar result has been reported for some sites in France.

In the U.K. a small but statistically significant excess risk of leukaemia, relative risk of 1.2, was detected around other nuclear facilities other than Sellafield. But a similar excess was also found around sites that had been selected for construction of nuclear power plants and never used. Clearly there remains some unanswered questions.

I would also return to studies of the health effects that may occur in populations exposed as a result of the Chernobyl accident in 1986.

As I suggested earlier, there are significant problems that will have to be overcome if these studies are to be successful.

First, however, the International Atomic

Energy Agency has already completed a comprehensive review reported in 1991 describing what information is likely to be derived for populations living outside the prohibited zone, an area roughly 30 kilometres surrounding the complex, and excluding those people which received the largest contamination.

Their conclusion was that radiation doses to these other populations were probably sufficiently small that long-term effects on their health are unlikely to be detected.

They also concluded that while there certainly are health effects measurable even now in these populations, they are unlikely to be directly related to their radiation exposures.

Now, with respect to the more highly exposed populations, there are generally two categories. First, a group of several hundred thousand workers, mainly army conscripts, brought in to help with clean up. These are known generally as the liquidators.

Although most of these workers have since returned to their homelands, my understanding is that records for many of them have been kept and that they have been entered into a program of regular medical checkups. These workers are the subject of study of

several groups, and particularly by a large U.S. led team of specialists to which Canada is contributing expertise.

The second group are the most highly exposed populations from the nearby villages who were evacuated soon after the accident. There are apparently some problems in tracing many of these people, but particular study populations from this group have been defined, for example, by the US/ Canadian team, focusing on the leukaemias and thyroid disease in children.

One of the very serious problems in conducting scientific studies of these populations is the lack of reliable health records for the period prior to the accident. Nevertheless, epidemiologists are hopeful that techniques can be developed to overcome these problems and that valid objective analysis can be completed, but it's clearly too soon to assess whether they will be successful.

So in summary, considerable effort has been and continues to be expended to relate exposure from radiation facilities to public health effects. At the present time there is no reason to believe that the present estimates of risk are grossly wrong.

Q. Finally, Dr. Whillans, do you see any

reason why Ontario Hydro cannot continue in the future to maintain and operates its facility to acceptable standards of occupational and public health?

A. From the perspective of health effects which arise as a result of routine operation and maintenance, I see no such evidence.

First I would like to point out that radiation is probably the most thoroughly studied environmental toxin. Despite residual uncertainties, there is a good understanding of both the acute affects and the long-term risk resulting from radiation exposure.

Second, this knowledge has led to the development of a sound system of controls which limits risk to workers and to the public. As a result, the average annual exposure of a radiation worker in an Ontario Hydro nuclear generating station, whether in routine operation or in rehabilitation work, is now probably less than he or she would receive from natural background.

Third, the exposures that do occur even to the most exposed members of the public as a result of routine emissions are only about 1 to 2 per cent of that background.

Finally Ontario Hydro's radiation control

program has been and continues to be good by international standards and the risks imposed are not dissimilar to those of any conventional safe industry.

Q. Thank you, Dr. Whillans.

Mr. Chairman, that concludes Dr. Whillans' evidence. I am not sure whether you would like to take a break now or start with Mr. King's evidence on nuclear safety.

THE CHAIRMAN: We will take the break now, bearing in mind that we are going to stop no later than a quarter to five. We will take the break now.

MS. HARVIE: All right.

THE REGISTRAR: Please come to order. This the hearing will recess for 15 minutes.

---Recess at 3:35 p.m.

---On resuming at 3:47 p.m.

THE REGISTRAR: Please come to order. This hearing is again in session. Be seated, please.

MS. HARVIE: Q. Starting now with you, Mr. King. Mr. King is going to be speaking to nuclear safety, as I mentioned earlier.

Mr. King, when you talk about nuclear safety what do you mean?

MR. KING: A. Well, the term "nuclear safety" is used in Ontario Hydro to refer to the

process of providing protection to the public following concern with radiological risks associated with Ontario Hydro's nuclear program. What I will be talking about in my evidence is the protection of the public from accidental releases of radiation.

Q. How can we be assured that nuclear power stations Ontario Hydro is operating are safe?

A. The short answer to that is, the hazard involved is well recognized, and there are systems in place to manage the associated risk from those hazards.

First of all, we have an independent, government-run regulatory body. We have in Ontario Hydro a comprehensive system of nuclear safety management. And thirdly, this whole process of nuclear safety management and regulation has been subject over the years to outside review with positive results.

Q. All right. Would you describe first how nuclear power is regulated in Canada?

A. Well, to start with, the legal basis for the regulation of atomic energy is the Federal Atomic Energy Control Act. This establishes the Atomic Energy Control Board, or the AECB, as the body which regulates nuclear power. In the federal hierarchical structure the president of the Atomic Energy Control

Board reports to the Minister of Energy, Mines and Resources.

The mission of the AECB is to ensure that the use of nuclear energy in Canada does not pose undue risk to health, safety, security or the environment. However, the AECB is not responsible for the safety of our nuclear power reactors in Canada. It's an important and fundamental concept in the Canadian approach to regulation that it is the operator that is responsible for safety. The role of the AECB is to set the rules and make sure that the holders of licenses follow those rules.

In order to set up a system of regulation the AECB has set up three steps. What you have to do is you have to get a site acceptance for a potential site for a nuclear power reactor, you have to get construction approval, and then operating license.

Once this process is complete an operating license can be issued from anywhere from one to five years after which the operating license has to be renewed.

The AECB is, I think, a very active and involved regulator. One of the ways they keep involved and informed is by having staff resident at all our stations to typically have one -- four to five staff

members resident at our stations, at each of our stations, and these AECB staff would regularly perform inspections, audits of our activities, and they prepare annual reports which look at our compliance with the regulations that they have established.

Sometimes these reports are fairly critical of Ontario Hydro's activities in certain areas, and we have to respond to these criticisms and make improvements.

Q. All right. Getting back to the AECB's licensing process could you describe what is required to obtain the various approvals and licenses, starting with site acceptance?

A. Well, to get site acceptance an applicant needs to establish first of all a conceptual design for the facility, describe that to the Atomic Energy Control Board, and it has to show the Atomic Energy Control Board that it is feasible to design, construct and operate that facility on that proposed site in order to meet all the applicable AECB regulations. An applicant would have to submit a site evaluation report as part of the process, and this later becomes part of the safety report for the station.

In this site evaluation report various

man-made and natural hazards that would affect the site would have to be studied; for example, seismic potential, any hazards involved with nearby industrial activities, pipelines, shipping, whatever. And you would have to show that these potential impacts would not affect the proposed facility or that impacts from them would have been taken into consideration in the design of the facility.

For example, in the seismic area there is a Canadian Standards Association standard, which defines the site and regional seismological investigations that have to be carried out to come up with a designation of the design-base earthquake for that site, and this would have to be done at the site acceptance stage.

If I could perhaps talk about a couple more examples, Darlington is the most recent site evaluation process that we have gone through even though it was quite a number of years ago, but in that instance the Darlington site has a chemical plant, the St. Mary's chemical plant which is on the next site over, and we would have gone to the company that runs that plant, look at all the hazards that could possibly evolve from that plant - they have a wharf, and they bring in ships, and what goes on those ships.

We would also look at any other shipping that goes by the lake at that point. There is a rail line that goes through the Darlington plant. We would have looked at all the toxic chemicals, explosive chemicals, and any other potential impacts from that rail line.

There are pipelines, natural gas pipelines that pass in that part of the country, and we would have looked at the size of those pipelines and the potential for explosions or vapour cloud explosions emanating from those pipelines.

There are other ones that we would have done at Darlington, but I think these give you the flavour of the sort of things we do in coming up with the site evaluation and the potential impacts from both man-made and natural phenomenon.

Q. And what about construction approvals?

A. When you get site acceptance you can do site preparation work on the site but you can't pour concrete. When you start pouring your base slab you have to have construction approval, and to get construction approval you would have to have the design in a fairly advanced state, such that the AECB could be assured that their requirements are going to be met.

The construction is only authorized after the design and safety analysis programs have progressed to the point where the AECB are convinced that none of the safety-related systems in the plant will need to be changed, the design will need to be changed following the issue of construction approval.

To apply for construction approval you will have to submit a preliminary safety analysis report. This would include the site evaluation report, which I mentioned before. It would also include a detailed design description of the facility as well as a volume on safety analysis, or many volumes on safety analysis, which would be conducted at that stage.

This safety analysis would involve the safety analysis of many postulated accidents that would have been established, design-basis accidents for that station.

There are many other requirements necessary to get construction approval which I haven't mentioned, but one of them would be to have an improved construction quality assurance program in place before you actually start any construction.

The safety analysis submitted at this stage would be very extensive. In fact, all accidents that could set any of the design parameters for any of

the safety systems would have to be analyzed and documented in the preliminary safety analysis report at this stage.

Q. And what about the final stage, the operating license?

A. Well, to get an operating license Ontario Hydro would have to show many things; first of all, that the plant as built now conforms to the design that was presented earlier; we would have to show that all the required operating procedures are in place. This goes from the normal startup and shutdown procedures for the plant, to operating manuals for every system in the plant, to abnormal incidents manual. This is the manual which has all the operator actions that are required following any postulated accident. That would have to be in place. As well, you would have to have completed the whole AECB approved commissioning program.

Now, the commissioning program is a program where all systems in the plant are tested out to make sure that they can perform their intended function in both the normal operations mode and any accident demands that are put on those systems.

We would also have had to have submitted a final safety analysis report. In this we would have

updated the design description if there had been any changes or updated the safety analysis part of the safety report if there had been any changes to that.

What we are doing at this stage is just making sure that that whole package of safety report conforms to the as-built design.

Also, to get an operating license we would have to have successfully completed the operator examinations and have received AECB authorization for senior operating personnel at the station.

We would also have to have in place an AECB approved document that we produce called "Operating Policies and Principles", and I will be talking about that a little more a little later.

We also have to have in place our emergency plans for the station, and again, I will be talking about that a little later as well.

One other thing I would like to mention, the same as at the construction approval stage, we have to have an approved quality assurance program, but this time an approved quality assurance program for operation.

There are many other requirements, as you may imagine, to get an operating license, but these are just some of the key ones that I would like to bring

out.

Also, when an operating license is granted there are conditions that come along with that license; for example, the total reactor power. You can't get any more power out of the plant than you are licensed for. You can't get any more power out of a particular channel than you are licensed for. You are not allowed to change the set points for the trip parameters for a reactor shutdown system without AECB approval.

You are not allowed to change really any of the design aspects of any of the safety-related systems without AECB approval. You are not allowed to change senior operating personnel or staff levels at the station without AECB approval.

What I am trying to say here is there are many conditions that go along with that license. It's not just once you get the license you operate like you want to operate.

Q. You have just described the regulatory process and its role in nuclear safety management. Could you now describe how nuclear safety is managed by Ontario Hydro?

A. Well, the primary objective of nuclear safety in Ontario Hydro is to prevent accidents

from occurring in the first place, and secondly, if they should occur to ensure that they will not lead to unacceptable consequences.

But before discussing this subject in detail I would first like to set up some sort of framework for this discussion, and I will be doing that by referring to nuclear safety management in both the design and operational phases of the station. To be effective, nuclear safety has to be managed properly in both of these phases.

But again, before continuing I would just like to mention that there is the detailed discussion of the nuclear safety management in Ontario Hydro in Exhibit 187, which is Ontario Hydro's submission to the Ontario Nuclear Safety Review.

In the design phase of a plant it is necessary to especially come up with the design that can be operated safely, and this is achieved by having a sound overall design concept and then by executing that concept through the use of the best available standards and design codes.

Also, in the design phase the safety adequacy of the design is demonstrated by analyzing various postulated accident conditions and showing that the resulting consequences, or any resulting

consequences, are within AECB allowed units.

To manage safety in the operational phase it is necessary to operate the plant within a defined set of safe operating parameters, and, as a further precaution, to have emergency plans in place in case an accident should occur.

I will be covering each of these three aspects - that is, managing safety in the design phase, in the operational phase, and the subject of emergency planning or preparedness in more detail later on in my testimony - but I would first like to mention two review groups which are pertinent to nuclear safety management in Ontario Hydro.

The first of these is the Nuclear Integrity Review Committee, or NIRC as it is called inside Ontario Hydro. This is a high-level internal committee. "High level", I mean it's composed of vice-presidents and director level type people.

Their role is to provide an ongoing overall assessment of the operational safety performance of our stations and safety design. This group would typically meet once a month to review the safety performance in the last period of time as well as any other important safety issues that would have arisen since their last meeting. The report on NIRC's

activities is tabled in the Ontario Legislature every year by the Minister of Energy.

The second review group that I would like to talk about is the Technical Advisory Panel on Nuclear Safety. This is an external review group. It is composed of -- on it it has international representation as well as -- it has primarily Canadian, but it has some international representation on it.

This panel was set up in response to a recommendation of the Ontario Nuclear Safety Review to advise the president of Ontario Hydro on the appropriateness, adequacy and quality of the safety aspects of Ontario Hydro's nuclear program.

Q. Mr. King, could you now describe the basic approach to reactor safety design?

A. Well, the basic approach to reactor safety design is one of defence and depth. This is a standard design practice when you are dealing with hazardous substances. Defence and depth means that you don't have a single line of defence against possible occurrences; you have many lines of defence.

[4:05 p.m.]

We recognize that there may be equipment or human failures over time and these are allowed for in the design.

Defence and depth results in plants being desired to such as there is a succession of physical barriers between the radioactive materials of concern and the public. These physical barriers are the fuel itself which has the potential to retain certain radionuclide species. The next barrier is the sheath that the fuel is in. If you remember the missing fuel bundle, it has these sheaths about the size of your pencil where the fuel is in. This zirconium metal sheath is a barrier which prevents fission products from leaving the fuel.

The heat transport system which is the system that the coolant, the heavy water coolant travels in to transport the heat from the fuel to the steam generator, the thick walled pipes of that heat transport system is another barrier, and the concrete containment that this whole reactor structure and heat transfer system is enclosed in is the final barrier.

Now, there are engineered safety features provided in the design to ensure that following any postulated design basis accidents, that these barriers remain sufficiently intact, that any releases are within allowable limits.

Q. Could you describe some of these engineered safety features, please.

A. Well, the four major engineered safety systems, we refer to these as special safety systems, are the two shutdown systems, the emergency coolant injection system, and the containment system.

These are describe in detail in each of the station safety reports, and I referred to earlier Exhibit 187, the Ontario Hydro submission to the Ontario Nuclear Safety Review, they are discussed in detail in that exhibit as well.

I believe the Darlington safety report has been submitted in response to Interrogatory 9.7.58, and hence the Darlington special safety systems would be discussed there.

But if I could start with the shutdown systems. All stations except Pickering "A" have two fully capable independent and diverse shutdown systems. One involves the dropping of neutron absorber rods into the core from above, and the other involves injecting a neutron absorbing solution into the moderator from the side of the reactor.

If I could draw your attention to the overhead which is up now, this is page 43 of your handout. Looking at my copy of the overhead, I will go through a description of it for you.

I direct your attention to the upper

right-hand portion of the overhead, you will see one box with a plus and a minus, that represents a power supply. I should mention here that this diagram is just a conceptual diagram, it doesn't represent accurately the number of components or it may be missing some components, but it just demonstrates the principal operation of this system.

There is the power supply in the upper right-hand box. You have a box labelled trip logic. Normally that power supply contact is made, the contacts in the trip logic box is made, such that there is a continuous electrical circuit.

The electromagnetic clutches are really wheels or pulleys, at each of the shutoff rods there is a cable connecting each shutoff rod, and that cable is wound around the pulley and that pulley is kept in place by electromagnetic clutch.

What happens, if you look at that trip logic box, we have sensors coming in. There are several trip parameters which would trip the reactor. This could be a high reactor coolant pressure, high reactor power, there are many of these parameters.

The three arrows represent three channels or trains of sensors for each parameter. When two out of three of those channels are above the trip set

point, which has been established for that parameter, it will break the electrical contact which will then release the clutch, the electromagnetic clutch and the rods will drop into the core.

You will also note that this design is fail-safe in that if you lose power supply to the first shutdown system, you will also get a dropping of rods.

This diagram shows three rods, in fact there is many more. Darlington has 32 rods, for example.

When you drop a neutron absorbing material, and these rods are made of cadmium, into the core, they absorb neutrons and by doing that the chain reaction fission process is stopped and the reactor is shut down.

If I direct your attention now to the bottom left-hand side of the diagram. The logic arrangement is very similar. I would just like to point out though that the sensors involved here are completely independent for the second shutdown system, this is the liquid injection system, from the first shutdown system, the rod drop system. They are independent in that they are different sensors completely and they are located in a different location.

The liquid poison system works -- and again while only one tank is represented here, in fact there are several tanks, and Darlington would have eight poison tanks.

When two out of three of the sensors detect that there should be a trip, this helium which is in this upper tank, it is under high pressure, it forces the gadolinium nitrite, which a neutron poison, a liquid, through the nozzle in the core and into the moderator. That poison absorbs neutrons and shuts down the reactor.

You will note that the first shutdown system comes in from above, the second shutdown system, the liquid poison system, comes in from the side of the core.

Now, Pickering "A" is different in that its two shutdown mechanisms are rod drop and moderator dump. It's the only reactor that has a moderator dump system.

Moderator dump, as the name suggests, is a dumping of the heavy water moderator which stops the fission reaction and shuts down the reactor.

Moderator dump at Pickering, at Pickering "A", is capable of shutting down the reactor fast enough for all accidents except a large loss of coolant

accident.

THE CHAIRMAN: I'm sorry, a large loss of?

MR. KING: A large loss of coolant accident.

A loss of coolant accident is where the heat transport system, which is the heavy water system conducting the heat from the fuel to the steam generator, if it has a rupture in it and you lose that coolant, that's referred to as a loss of coolant accident.

It's only for the largest of these, which means the largest rupture of the largest pipe, that the moderator dump is not effective in shutting down the reactor.

For smaller loss of coolant accidents, loss of regulation type of accidents, losses of control of reactor power, the moderator dump is effective.

Enhancements to the Pickering "A" shutdown systems have been proposed to the AECB recently and these are currently under review and discussion with them.

If I could move on to the emergency coolant injection system now. These are provided at each station and these systems are to inject water and

ensure cooling of the fuel in these loss of coolant accidents that I just referred you to.

Now, if I can direct your attention to the overhead, this is on page 44 of your handout. On the left-hand side of the figure there is the representation of the main reactor systems. In the middle of the left-hand side, the various pipe work there would be the heat, main heat transport system.

If you have a rupture in that piping then you need to provide water into the heat transport system to maintain the cooling of the fuel. This is provided by the system on the right-hand side of the diagram.

This particular system, the design of it is a wee bit different from station to station. This particular system here represents the one installed at Bruce "A" and "B".

The ECI system, emergency coolant injection, is composed of three timed phases. There is the early high pressure phase which is represented by the two tanks in the upper right-hand part of the diagram. What happens here is that this valve at the very top right is normally closed. It separates high pressure nitrogen, which is in the accumulator tanks shown on the diagram, from light water, normal water in

the left-hand tanks. Again, this is just a conceptual diagram, there are more tanks than shown here.

So when the system detects there is a need for injection, this valve will open, the high pressure nitrogen will force the light water through the valving into the heat transport system.

Once this tank of light water is exhausted, then you go to the next time phase.

Here, in the bottom right-hand part of the diagram, you will see a tank labeled grade level water tank, and you will note there are some pumps called ECI recovery pumps as well.

What happens after the high pressure tanks are exhausted, these recovery pumps take water from the grade water level tank, inject it into the heat transport system for another period of time. And the period of time depends on how large the break in the heat transport system is postulated to be.

The long-term phase involves this recovery sump, which is in the middle lower part of the diagram. The water leaving the break in the heat transport system flows to the basement, the floor of the reactor building. It's collected in this sump, and the recovery pumps can draw water from that sump and complete the circuit, and that complete circuit is

maintained for weeks or months, or whatever length of time it's required.

MS. HARVIE: Q. Thank you. Could you explain what activities go on in the design phase to verify that reactor design is adequately safe?

MR. KING: A. Before I do that, perhaps I could continue and discuss the containment system.

The containment system is really made up of a set of sub systems, and they ensure that the containment boundary remains intact following all postulated accidents. It does this through the isolation of any penetrations which are normally in the open position in the containment envelope, as well there is a large vacuum building which is an integral part of Ontario Hydro's approach to containment. This vacuum building ensures that the containment pressure remain sub atmospheric for an extended time following postulated failures.

You will note on this next overhead, which is on page 42 of your handout, an illustration of containment.

On the right-hand side of this diagram is the vacuum building. As the name suggests, it is at almost a pure vacuum, all the air, most of the air has been exhausted from that building.

It's a very large building. If you recall Mr. Penn's photos of the Bruce or Pickering site, it was the largest building on the site.

It contains a dousing system. This system works through passive means. There are no moving parts. It works purely on pressure differential.

In fact, there are two vacuums in the vacuum building, there is the main chamber vacuum and right under the letters i-n-g in dousing is a little raised part, that has an upper chamber vacuum which is separate from the main chamber vacuum.

If the pressure in the main chamber vacuum increases just through the differential pressure principles, it will force water down through the spray nozzles, and that's basically how the vacuum building works.

In the middle of the diagram there is a device called a self-actuated relief valve. This again does not require any outside services, electrical power, whatever to work; it works purely on pressure differentials. It has weights on it which keeps it in the closed position.

On the left-hand side of the diagram is a representation of a reactor building. The circle is a

simple representation of a heat transport system, and the diagram shows a break in that piece of pipe.

So what would happen following a loss of coolant accident. There would be a release of high pressure, high temperature steam, and liquid mixture in the reactor building. It would entrain some air with it. This self-actuated relief valve would then open through differences in pressure on each side of that valve. The hot mixture would be drawn into the vacuum building, raise the pressure, initiate the dousing, and then condense the steam, cool the mixture, and the whole atmosphere would then be retained in a sub atmospheric state for a period of days or weeks, and that ensures that any radionuclides that are present don't get out. In fact, air leaks in if there are any small impairments in the envelope.

These four systems I have talked about, their design does not remain statistic throughout the whole life of the plant. They are occasionally upgraded when necessary. For example, the Pickering "A" and Bruce "A" and "B" ECI systems, emergency coolant injection systems, have had substantial design upgrades over the years, and as I mentioned a little earlier, the Pickering "A" shutdown systems, there are currently enhancements in the final stages of

resolution for the Pickering "A" station.

Q. I'm sorry I cut you off last time.

Now, would you explain what activities go on in the design phase to verify that reactor design is adequately safe?

A. The purpose of safety verification is to demonstrate that the plant design is such that the risk to the public is sufficiently low. This is done by showing that the AECB regulatory requirements have been met.

Now, the primary safety verification vehicle is safety assessment, where safety assessment is made up of two components. First of all, there is the traditional or deterministic safety analysis, and secondly, there is probabilistic safety assessment.

Q. All right. Would you please describe the traditional or deterministic safety analysis?

A. Well, the deterministic safety analysis is the process of predicting consequences of a large set of design basis accidents and comparing these consequences to AECB limits. And as I have mentioned earlier, the results of these analyses are reported in the safety reports for each station.

I have two overheads which show the AECB applicable limits. This one here, which is on page 41

of your handout, these are the limits which apply to all reactors except Darlington. These are commonly referred to as the sighting guide limits or the single dual failure limits.

The AECB have established two categories of postulated accident, the first category being a single failure, the other category being a dual failure.

The first row in this figure has the limits for a single failure. A single failure is the failure of one of the systems which are required in the normal production of power from the reactor. And as you can see, that there are individual dose limits, dose limits for the individual, and a population dose limit as well, collective dose limit.

The second row has the limits for dual failures. These are single failures, failures of the process systems in the plant combined with the failure of any of the special safety systems which I have just discussed. And these limits, these accidents which would be much more unlikely, their limits are in the second and third columns.

Now, if I can turn your attention to the next overhead which is on page 40 of your handout. These are the limits that apply to Darlington.

When Darlington was being licenced approximately in the 1980 time frame, the AECB came out with a new consultative document which established some new rules for safety analyses, and these were used on a trial basis on Darlington.

[4:26 p.m.]

What the AECB established is the left-hand column and the two right-hand columns

They divided the likelihood range, the frequency range of possible accidents into five categories rather than the two that we had in the siting guide, and they established in this document various accidents in each of these classes that we would have to consider.

We would also have to, based on the design that we were proposing, come up and identify other accidents that would be appropriate to include in the set of design-basis accidents.

And in the right-hand columns are the individual dose limits for the various categories.

This document did not establish, this C6 consultative document, AECB document, did not establish population dose limits, but in the licensing of Darlington these were calculated and documented in the safety report.

I guess there is one other thing I would like to point out on both of these, and this is the bottom of the column labeled "Whole Body Dose Limit". The largest limit allowed, the largest dose allowed is .25 sieverts, and if you will recall Dr. Whillans' testimony that this would be in the stochastic range that means that the highest allowed dose still in the stochastic range would mean that there is no detectable health effect, no observable health effect at that dose level. And that's the same for both these limits and as well the siting guide limits in the previous overhead.

Now, the set of design-basis accidents that we would have to go through and analyse and compare to these limits, there is a very long list.

Just giving you an example of some of them, there would be losses of coolant which I have gone through before, but there are various types of losses of coolants.

There are the ruptures in the main, large piping, typically in the, oh, 12 inches to 20 inch type range diameter piping, to the feeder pipes which are in the 2 1/2 to four inch diameter range, to the pressure tubes which are normally four inches in diameter. Loss coolant accidents in the steam generator tubing would

also be included.

There is a large range of loss of coolant accidents, depending on site and location.

There also have to be in this design-basis set accidents involving losses of nuclear power regulation, increases of reactivity of power in the core at various rates.

We would have to look at losses of steam generator feed water, as well as failures of the piping which brings feed water to the steam generators, as well as steam -- failures of steam piping that leave the top of the steam generators.

We would also have to look at losses of moderator cooling; fuel channel blockages - each of the fuel channels in the reactors, we have to assume that there is a blockage in that and it gets no heavy water coolant; the various losses of electrical power. There are various electrical power systems in the plant.

Each of these in turn would have to be assumed to fail as well as for the service water systems which provide cooling and instrument air systems which provide control air to various systems. We have to assume that these fail and...

As well, we would have to look at all the external events in the design-basis set. These are the

earthquakes, any other external hazards which I have discussed earlier when talking about the site evaluation process, as well as internal plant flooding, internal fires.

These are all the single events, and then of course we would have to go through the whole set of dual failure analysis, taking each of these single-process failures and combining them in turn with failures of the four special safety systems.

In doing all this analysis it would usually require detailed computer models of the plant and all the associated phenomena that could transpire during these accident progressions.

These computer models would previously have to have been verified against experiments, which we would conduct as part of our safety research program.

We have a safety research program, which is conducted with AECL, which is conducted with our other sister utilities in Canada, and as well we are involved in several international collaborative efforts to conduct experiments. It is with these experiments that we can verify the codes that we use in our safety analysis, our computer codes.

Once the safety analysis is complete,

though, it just doesn't stay static. We have a requirement to update the safety analyses through the lives of the stations, and, as I mentioned earlier, the Darlington safety report has been tabled as an exhibit or, in fact, in response to Interrogatory 9.7.58. I think it is probably about seven volumes, seven binders of safety analysis, which we would have submitted to the control board at Darlington.

Q. In addition to the deterministic safety analysis you have also mentioned probabilistic safety assessment. Would you now please describe this probabilistic safety assessment?

A. The probabilistic safety assessment, as I mentioned, is also performed as part of the safety design verification activities.

Some people may have heard this referred to as probabilistic risk analysis or PRAs or PSAs. Internationally it has picked up a large number of possible terms.

The one that I will be using is PSA, which stands for Probabilistic Safety Assessment, which is the more generic term, if you will.

When you are doing deterministic safety analysis, now the object was to show that for the defined set of accidents consequences are less than

AECB limits, but there are normally a set of conservative assumptions which you have to follow in conducting that analysis.

For example, the reactor regulating system, which has the capability of shutting down the reactor, we aren't allowed to take credit for that system in shutting down the reactor in our deterministic safety analysis. We can only take credit for the two shutdown systems, or actually, only one of them at any particular point in time.

But in probabilistic safety analysis the object is slightly different. It is to define what accidents occurring at what frequencies can lead to various defined levels of consequence, and to make that meaningful you have to do that analysis using best estimates or assumptions in order to get the right representation of the frequency.

Now, PSA makes use, extensive use of the various tools of reliability engineering, of event trees, fault trees, and it has some unique characteristics which augment the deterministic safety analysis which has been performed.

Over in the world over the last 10 years it has become standard practice that PSAs are performed on reactors in most countries.

In Canada in the late 1970s, early 1980s, we performed PSAs of an earlier vintage methodology on Bruce "A" and "B" and Pickering "B", and in 1987 we published a large study which uses current PSA methodology on Darlington.

Currently, we have a program of preparing current methodology PSAs on all reactors, and these are being done right now or will be done in the upcoming years.

Through the use of PSA you can identify those accidents which dominate the risk from the station, and by doing so you can identify areas for preferential improvement.

The Darlington PSA, which is also known as the DPSE or Darlington Probabilistic Safety Evaluation, was submitted in response to Interrogatory 9.7.12. This report and the other ones coming up are very large documents. The Darlington PSA is composed of around 12,000 pages of report that fit into about 20 volumes and six to eight feet of bookshelf.

MS. HARVIE: I think we have time for one more question, Mr. Chairman.

Q. You have just described, Mr. King, how Ontario Hydro manages safety in the design phase. Would you now describe how safety is ensured in reactor

operation?

MR. KING: A. Well, first of all, it is the individual station managers at each station who are responsible for the safe operation of their stations.

Meeting this responsibility involves, among others, five key elements which I would like to talk about.

These are that key personnel are well trained and have defined responsibilities and limits to authority; secondly, that there is a defined safe operating envelope for the station; that all work activities at the station are documented and approved before they are carried out; that regular surveillance of the status of equipment and systems is performed regularly; and that any operational occurrences that do occur are evaluated, their root causes determined, and any required corrections made.

Q. Well, we have six more minutes. We might as well start into the next question, take advantage of the time that we have.

Would you discuss these five elements in more detail, Mr. King?

A. With respect to the first element, following appropriate training and selection key personnel are approved by the AECB, as I mentioned

earlier.

These personnel include the station manager, the technical manager at the station, production manager, the senior health physicist at the station, the various shift supervisors, and each station would run five shifts, the shift operating supervisors and the unit first operators. These are the individuals who are in front of the panel and in control of the reactor at any time. These are called unit first operators.

The shift supervisors, the shift operating supervisors and the unit first operators, these are licensed AECB positions, and to obtain a license these personnel must pass a set of examinations. These examinations are both written and they are also examined on the full-scope control room simulator which exists for each station.

With respect to the second element there is a document called "Operating Policies and Principles" or OP&Ps as again they are referred to in Ontario Hydro. These exist for each station and are approved by the AECB.

This document achieves several purposes. Firstly, it defines responsibilities and safe operating policies. It says who can do what and what they should

do. It sets out administrative limits to authority, what people aren't allowed to do. It establishes a safe operating envelope for the station, and it defines actions to return to the safe state if you find yourself outside the bounds of the safe operating envelope.

Now, the safe operating envelope is really a set of numeric conditions, whether it's pressures, temperatures, levels in tanks, which defines a preanalyzed condition which would have been analyzed in the safety report which would have been shown to be safe. The safe operating envelope also defines equipment configurations, how many pumps in a certain system would have to be available before you are within the safe operating envelope, for example.

The third element of safe operation is the existence of a formal work authorization process to ensure that it is safe to do any maintenance, any repair, any testing of systems before this work is carried out.

This is to ensure that the work is carried out within the rules of the OP&Ps, the Operating Policies and Principles, and hence that the operation within the safe operating envelope is maintained.

With respect to the fourth element, there is a comprehensive operational surveillance program in place, and an operational surveillance program is composed of two sub elements. There is a component in-service inspection program and a system operational reliability monitoring program.

Now, the main purpose of the in-service inspection program is to regularly test pressure-retaining components in the plant: your pressure-retaining vessels, pressure-retaining piping, welds in piping, pressure tubes.

Throughout the life of the plant these non-destructive tests are made of these components to assure that there has been no degradation and that they still maintain their pressure-retaining capability.

The second subpart of the operational surveillance program is the operational reliability monitoring program. Here we are looking at systems and testing these safety systems to confirm that the availability of the systems is where we want it and they will be available if called upon to perform their safety function.

As a result of this testing, if there are any failures detected these would have to be recorded, and there are established AECB reliability requirements

which would have to be shown to be met.

Finally, with respect to the fifth element of safe operation, any operational occurrences that do occur in the lives of the station are studied under a system we call the Significant Event Reporting System. Here the SERs, or significant event reports, are written for any event of potential safety significance, but they are also written for events of significance with respect to occupational dose, occupational safety, production as well as environmental concerns.

As I mentioned earlier, these events are investigated, try to determine the root cause, extract any lessons learned from that experience, and make any modifications whether the modifications are in systems or procedures that are required.

Currently, we would produce around 600 to 700 significant event reports in a year for all of our units. That's not each, but altogether. And since 1980 there have been around 6,600 of these significant event reports written up.

Now, this doesn't mean that there has been 6,600 accidents by any means. I think what it shows is that the threshold for reporting is fairly low, but we want to pick up those precursors to more

important events, study those, and make sure that if you can control the precursors you can control the major event that might occur with the number of events occurring in a sequence.

MS. HARVIE: All right. I think that is enough for today, Mr. Chairman.

THE CHAIRMAN: We will adjourn then until tomorrow morning at ten o'clock. Do you know how much longer you expect to be, Ms. Harvie?

MS. HARVIE: We will not be any later than 12:00, 12:30 tomorrow I don't think at least.

THE CHAIRMAN: And then we have the motion by Energy Probe; is that right?

MS. HARVIE: Yes, that is correct.

THE CHAIRMAN: And then start cross-examination?

MS. HARVIE: Yes.

THE CHAIRMAN: Okay.

THE REGISTRAR: Please come to order. This hearing will adjourn until ten o'clock tomorrow morning.

---Whereupon the hearing was adjourned at 4:46 p.m. to be reconvened at 10:00 a.m. on Wednesday, March 25th, 1992.

E R R A T A
and
C H A N G E S

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Date: Tuesday, March 10th, 1992.

Page No.	Line No.	Discrepancy
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21005	25	bait s/r undertaking